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Block Valve, EC 90261
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Block Valve, EC 90261

Please find enclosed a review draft of S-110, Radiation Effects Analysis of Structural Support Block Valve, EC 90261. Any comments, corrections, additions or deletions would be appreciated.

W. E. French
W. E. French
Radiation Effects

MASTER

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TABLE OF CONTENTS

	PAGE
I. SUMMARY AND CONCLUSIONS	1
II. INTRODUCTION	2-3
III. ENVIRONMENTAL REQUIREMENTS	3
IV. DISCUSSION	3
A. 6061-T6 ALUMINUM	4
B. 440C	5
C. 347 STAINLESS STEEL	6-7
D. PHOSPHOR BRONZE	8
E. 321 STAINLESS STEEL	9
F. A-286 STAINLESS STEEL	10-11
TABLE 1 - SUBASSEMBLIES	12
TABLE 2 - AL 6061-T6 - TEST SUMMARY NO. & DATA POINTS	13-14
TABLE 3 - EFFECT OF IRRADIATION ON TENSILE PROPERTIES	15
TABLE 4 - 347 STAINLESS STEEL - TEST SUMMARY NO. & DATA POINTS	16
TABLE 5 - EFFECT OF IRRADIATION ON TENSILE PROPERTIES	17
TABLE 6 - A-286 TEST SUMMARY- NO. OF DATA POINTS	18
TABLE 7 - EFFECT OF IRRADIATION ON TENSILE PROPERTIES	19
REFERENCES	20
REIR 631-321 STAINLESS STEEL	21
REIR 632 -PHOSPHOR BRONZE	22

I. SUMMARY AND CONCLUSIONS

An analysis of the availability of data for candidate structure support block valve materials (aluminum 6061-T6, stainless steel 440C, stainless steel 347, phosphor bronze, stainless steel 321 and stainless steel A-286) with respect to the anticipated nuclear environment and the properties required for design has been completed.

Previous reactor tests have supplied information on tensile properties within the general fluence/temperature range of the NERVA requirements for the SSBV candidate materials except for phosphor bronze and 321 stainless steel. The tests are small in scope and served to provide answers for particular situations. None of the tests provide sufficient experience (reliability and confidence level) comparable to the properties and fluences dictated by the presently specified NERVA engine useful life which is 10 hours.

Aluminum 6061-T6 and stainless steels 347 and A-286 exhibit increased tensile strength and decreased ductility at SSBV fluences and temperatures. These materials are ductile under SSBV usage. No further material characterization tests are required.

321 Stainless steel is expected to exhibit the same characteristics (increased tensile strength and decreased ductility) as 347 stainless steel. Because of its critical use (bellows seal) and lack of data, a tensile screening test, REIR #631, has been prepared.

Irradiation damage of phosphor bronze while is a candidate material the valve seal is not known. A tensile screening test, REIR #632, has been prepared.

II. INTRODUCTION

The purpose of this S-110 Radiation Effects Analysis is to investigate and document the effects of nuclear radiation on the materials and components of the Structural Support Block Valve EC-90261, as a portion of the Propellant Feed System EC-90218. No evaluation of the SSBV assembly electro-mechanical actuator mechanism and converter-controller-amplifier are included in this analysis. The materials used in the SSBV are evaluated to determine if significant mechanical/physical property changes can be expected to occur when subjected to the NERVA anticipated thermal and radiation environments. This evaluation was conducted by examining data from NERVA program and other nuclear programs.

Where there is insufficient data available to make the proper evaluation, radiation effects information requests (REIRS) are prepared and define what data is needed to support the design and analysis requirements.

This SSBV "chapter" of the S-110 contains the following information:

1. A brief description of the SSBV, its location and an identification of each material of its major subassemblies.
2. A definition of the environmental and operational requirements for each material.
3. An analysis of irradiated material property data for each material.
4. Conclusions regarding the acceptability of each material for the intended applications ^{and} on the need for additional testing.
5. A compilation of supporting data for each material.
6. Radiation Effects Information Requests for specific irradiation materials properties data needed for the SSBV.

The SSBV assembly EC-90261 is used in the Propellant Feed Subsystem EC-90218, to permit propellant flow through the structural support stem coolant flow path during all engine operating modes except during the engine cooldown operation. It is also used to permit or prevent propellant flow through the pump discharge line during the engine cooldown operation.

II. INTRODUCTION (continued)

The SSBV assembly remains in the closed position during all engine non-operating modes and remains in the open position during engine pre-start conditioning.

During the engine cooldown operation the SSBV assembly is employed to accommodate pulse cooling of the reflector assembly. The SSBV assembly in the closed position blocks propellant flow in the event the Structural Support Coolant Valve (SSCV) becomes inoperative during any engine operating period.

SSBVs are located approximately at station 35.50 and ^{at} station 26.65. Both SSBVs are approximately at R 25.75 (Ref. 1).

There are ten major components in the SSBV. The materials have been selected for these radiation resistance at cryogenic temperatures in a hydrogen atmosphere. The major subassemblies of the SSBV and the materials used in the subassemblies are shown in Table 1.

III. ENVIRONMENTAL REQUIREMENTS

The extreme environmental conditions are listed below (10-hour nominal rated power) based on a graphite cone (Ref. 1):

Nuclear, n/cm^2 ($E > 1.0$ Mev)	1.5×10^{16}
n/cm^2 ($E < 0.4$ ev)	2.7×10^{16}
Gamma, Kerma Rads	5.5×10^7
Temperature, °R	48
Pressure, PSIA	1433
Liquid or Gaseous Hydrogen	

IV. DISCUSSION

Each of the materials listed in Table 1 Major Subassemblies is analyzed in this section for their suitability in the expected radiation environment.

A. 6061-T6 ALUMINUM

Aluminum 6061-T6 is an aluminum-magnesium-silicon alloy which is hardened by thermal treatment through age hardening after solution heat treating.

6061-T6 forging is the prime material for SSBV valve body.

The 6061-T6 test experience includes tensile tests of ring and pressure vessel forging (GTR-19, Specification QQ-A-327B) and plate (GTR-19, GTR-16 and Plumbrook, Specification QQ-A-327B). These tests include twenty-four irradiated specimens at 5×10^{16} to 1×10^{18} n/cm² ($E > 1.0$ Mev), tested at 30-40°R and 140°R.

Table 2 is a summary of data points and average percentage elongation.

Table 3 is a tabulation of the Effect of Irradiation on Tensile Properties of Aluminum 6061-T6. Elongation of 6061-T6 plate is not changed, 24% for control and 24% for the specimens irradiated in liquid hydrogen to a fluence of 5×10^{16} n/cm² (fast) and tested in liquid hydrogen (Ref. 3). Yield strength increased from 47 to 55 ksi for the unirradiated and irradiated specimens, respectively. Tensile testing (Ref. 4) of 6061-T6 plate after irradiation in liquid hydrogen to a fluence of 3×10^{17} n/cm² (fast) and testing in liquid hydrogen resulted in a decrease in elongation (i.e., 26% unirradiated and 18% irradiated) accompanied by an increase in yield of 53 ksi for the unirradiated and 72 ksi for the irradiated specimens.

Additional tensile testing (Ref. 5) of 6061-T6 plate after irradiation in helium to a fluence of 10^{17} n/cm² (fast) at 30°R, resulted no change in elongation (30%) accompanied by an increase in yield of 50 ksi for the unirradiated and 59 ksi for the irradiated specimens. Further testing (Ref. A) of ring and pressure vessel forgings in liquid nitrogen confirms the above results.

Aluminum 6061-T6 appears to be ductile over the temperatures and fluences in the SSBV valve body. At all fluences and test temperatures 6061-T6 show increased tensile strength. 6061-T6 is a suitable material for use in the NERVA SSBV in the radiation environment of 10^{17} n/cm² (fast) with liquid and gaseous hydrogen in the predicted temperature of 40° to 540°R. No additional radiation effects testing is required and no REIRS have been issued.

B. 440C

Material 440C is the prime candidate for fabrication of the rolling elements (ball bearings). For this use, 440C (a high-chromium high-carbon grade) is heat treated to a fully hardened condition ($R_c 56$ to $R_c 60$) and is martensitic. The material is brittle at room temperature and the ductility, as measured by elongation is nil. Both cryogenic and nuclear - cryogenic environments appear to increase the brittle behavior.

Irradiation (Ref (5)) at 30°R in helium to a fluence of $1 \times 10^{17} \text{ n/cm}^2$ ($E > 0.5 \text{ MeV}$) resulted in a decrease in ultimate strength. The ultimate tensile strength is 261 ksi for the unirradiated and 218 ksi for the irradiated specimens. The ductility (elongation) is nil for the unirradiated and for the irradiated specimens. Both irradiated and unirradiated specimens failed at less than 0.2% plastic strain.

The effects of high-pressure (10,000 psi) hydrogen at room temperature on 440C was studied at Rocketdyne (Ref 6). Although there is extensive scatter in the tensile data, it can be concluded that the embrittling effect of hydrogen on 440C at room temperature is extensive. The ultimate tensile strength of in 10,000 - psi helium is 293 ksi; and in 10,000 - psi hydrogen, the ultimate ~~tensile strength~~ is 119 ksi. Ductility (elongation) is nil in both conditions. Notch sensitivity of the 440C is extensive. Notch tensile strength in helium is 149 ksi and in hydrogen 74 ksi.

Tensile testing of this material indicates "brittle fracture" under cryogenic conditions with specimens unirradiated or irradiated. However, for bearing applications in cryogenic conditions 440C is used extensively and qualifications of bearing assemblies are based on "use tests". Bearing "use tests" are planned for the NERVA TPA Bearings. No REIRS have been issued for bearing use tests in the SSBV.

C. 347 Stainless Steel

AISI Type 347 stainless steel is a 18-8 chromium-nickel austenitic stainless steel. The structure of this steel is, essentially, a interstitial solid solution of carbon in highly alloyed gamma iron.

347 stainless steel is the prime material for the valve ball, ball support assembly, seal lift assembly and double slider coupling. The 347 test experience includes tensile tests of plate, welded and non-welded (GTR-19, Specification AMS 5512 annealed), wrought sheet (GTR-16, Specification MIL-S-6721B annealed), sand casting (GTR-16, Specification, AMS 5363B annealed), round bar (Plumbrook, ASTM-A-167-158 annealed) and forging (GTR-20C; Specification QQ-S-763C). These tests include fifty-two irradiated specimens at 5×10^{16} to 6×10^{18} n/cm² ($E > 1.0$ MeV), tested at 40°R and 140°R. Eight of the specimens were welded plate and thirteen of the specimens were annealed to 540°R prior to tensile testing. Table 4 is a summary of data points and average percentage elongation. Table 5 is a tabulation of the effect of irradiation on the tensile properties of 347 stainless steel.

Under cryogenic condition (liquid hydrogen and liquid nitrogen) and irradiated to a fluence of 10^{17} n/cm² (fast) the property change in stainless steels is an increase in yield strength and ultimate strength. These changes in properties are usually accompanied by a loss in ductility (Ref. 7).

Elongation of 347 plate is decreased from 27.2% from the control to 26.2% for the specimens irradiated in liquid hydrogen to a fluence of 3×10^{17} n/cm² (fast) and tested in liquid hydrogen (Ref. 4). Yield strength increased from 65.6 ksi to 87.3 ksi for the unirradiated and irradiated specimens, respectively. Elongation of 347 welded plate (Ref. 4) is decreased from 21.9% for the control to 19.7% for the specimens irradiated in liquid hydrogen

to a fluence of 3×10^{17} n/cm² (fast) and tested in liquid hydrogen. Yield strength increased from 67.1 ksi to 90.0 ksi for the unirradiated and irradiated specimens, respectively. Additional tensile testing (Ref. 5) of 347 round rod after irradiation in helium to a fluence of 10^{17} n/cm² (fast) at 30°R and testing at 30°R, resulted in a decrease in elongation (i.e., 41.3% unirradiated and 37.0% irradiated) accompanied by an increase in yield strength of 51.9 ksi for the unirradiated and 64.0 ksi for the irradiated specimens. Further irradiation and testing of 347 plate (Ref 4) and forging (Ref. 8) in liquid nitrogen confirms the results above.

The effects of high pressure gaseous hydrogen on 347 stainless steel in the predicted radiation environment has not been determined. Studies by W. T. Chandler at Rocketdyne (Ref. 6) on austenitic stainless steels (310 and 316) indicate slight or negligible embrittlement due to high-pressure hydrogen (10,000 psi) at room temperature. Notch strength of 310 SS decreased 3% in 10,000 psi hydrogen and there was no change in elongation on unnotched specimens in the hydrogen atmosphere. There was no change in 10,000 psi hydrogen atmosphere notch strength and elongation of unnotched specimens of 316SS. Austenitic 347 stainless steel would be expected to exhibit slight or negligible embrittlement due to high pressure hydrogen.

347SS is suitable for SSBV components in the predicted NERVA environment (40 to 540R, hydrogen atmosphere and 10^{17} n/cm² (fast)). Further study of this material with regards to the specific NERVA applications to upgrade the data category is necessary. Additionally, the synergetic effects of stress/hydrogen and radiation should be investigated if this effect is determined to be significant in similar austenitic stainless steels. No REIRS have been issued for this material.

D. Phosphor Bronze

Phosphor bronze is the prime candidate material for the seal in the NERVA SSBV.

✓ This material is

No radiation effects data are available on this composition (See REIC Report No. 45).

✓ Preliminary screening for irradiated material properties (tensile and fracture toughness) with regards to the specific NERVA applications is necessary for this material. Additionally, the synergetic effects of stress/hydrogen and radiation should be investigated. REIR #632 has been prepared to determine the characteristics of this material in the expected NERVA environment.

E. 321 Stainless Steel

AISI Type 321 stainless steel is a chromium - nickel stabilized austenitic stainless steel. 321 SS is the prime material for the bellows seal and the static seal (lead coated) in the SSBV.

No irradiation data are available for 321 SS (See REIC Report No. 45).

Irradiation of 347 SS (a stainless steel similar to 321 stainless steel) in liquid hydrogen to a fluence of 3×10^{17} n/cm² (fast) and testing at cryogenic conditions reveal slight changes in tensile properties (Ref. 4). The yield strength increased from 65.6 to 87.3 ksi after irradiation and the elongation was 27.7% unirradiated and 26.6% after irradiation. Other irradiation tests of 347SS (Ref. 5) at a fluence of 5×10^{16} n/cm² (fast) in liquid hydrogen show an elongation of 40.2% for unirradiated and 41.0% for irradiated specimens. Austenitic 321 stainless steel would be expected to exhibit similar tensile properties as 347 stainless steel.

The effects of high-pressure gaseous hydrogen on 321 stainless steel in the predicted radiation environment has not been determined. Studies by W. T. Chandler at Rocketdyne (Ref. 6) on austenitic stainless steels (310 and 316) indicate slight or negligible embrittlement due to high-pressure hydrogen (10,000 psi) at room temperature. Austenitic 321 stainless steel would be expected to exhibit similar slight or negligible embrittlement due to high pressure hydrogen.

The use of 321 stainless steel for the SSBV in the predicted environment appears to be suitable considering the radiation involvement. However, 321 stainless steel will be used as a bellows seals subassembly which are in the failure mode, as determined by the failure mode analysis. Initial tensile testing of this material with regards to the specific NERVA applications is necessary. Additionally, the synergistic effect of stress/hydrogen and radiation should be investigated if this effect is determined to be significant in similar austenitic stainless steels. REIR #631 has been prepared to determine the characteristics of this material in the expected NERVA environment.

F. A-286 Stainless Steel

A-286 is a precipitation - hardening, austenitic, iron-nickel-chromium alloy designed for service at extreme variation of temperatures where high strength and good corrosion resistance are required. This alloy is commonly used in two different heat treated conditions, annealed at 1800F prior to aging. The materials and heat treatments conform to AMS Specifications 5735D and 5737B respectively.

A-286 cold work and aged is the prime material for valve bolts, nuts and screws. A-286 forging is the back-up material for valve bodies. The A-286 test experience includes tensile tests of forging (GTR-19, Specification 5735), plate (GTR-16, Specification AMS-5525), Rod (Plumbrook, AMS 5735 and AMS 5737) and bolt (GTR-20C, AMS 5735 cold worked). These tests include fifty irradiated specimens at 1×10^{17} to 6×10^{18} n/cm² ($E > 1.0$ MeV), tested at 40°R and 140°R. Fifteen of the specimens were annealed to 540°R prior to tensile testing.

Table 6 is a summary of data points and average percentage elongation.

Table 7 is a tabulation of the effect of irradiation on tensile properties of A-286 stainless steel.

Under cryogenic condition (liquid hydrogen and liquid nitrogen) and irradiated to a fluence of 10^{17} n/cm² (fast), the property change in stainless steels is an increase in yield strength and ultimate strength. These changes in properties are usually accompanied by a loss in ductility (Ref. 7).

Elongation of A-286 disc forging (Specification AMS 5735) is decreased from 25.8% for the control to 23.0% for the specimens irradiated in liquid hydrogen to a fluence of 3×10^{17} n/cm² (fast) and tested in liquid hydrogen (Ref. 4). Yield strength increased from 146.8 ksi to 156.5 ksi for the unirradiated and irradiated specimens, respectively. Additional tensile testing (Ref. 5) of A-286 Round Rod (Specification AMS 5735), after irradiation in helium to a

fluence of 10^{17} n/cm² (fast) at 30°R and testing at 30°R, resulted in a slight decrease in elongation (i.e., 34.5% unirradiated and 33.5% irradiated) accompanied by an increase in yield ^{strength} of 149.6 ksi for the unirradiated and 152 ksi for the irradiated specimens. Further irradiation and testing of A-286 disc forging in liquid nitrogen (Ref. 4) confirms the results above.

✓ Elongation of A-286 Bolts (Specification AMS 5735 cold worked) irradiated and tested in liquid nitrogen (Reg. 9) decreased from 16% ~~from~~ the control to 15% irradiated to 4.3×10^{17} , 13% irradiated to 1×10^{18} and 10% irradiated to 6.5×10^{18} n/cm² (fast). Yield strength increased from 223 ksi to 247 ksi for the control and irradiated to 6.5×10^{18} n/cm² (fast) specimens, respectively.

✓ The effects of high-pressure hydrogen (10,000 psi) at room temperature on A-286 stainless steel have been studied by W. T. Chandler at Rocketdyne (Ref (6)). Results of tensile testing both notched and unnotched specimens in helium and hydrogen conclusively show that A-286 stainless steel is virtually unaffected by high-pressure hydrogen. Elongation was 26% in helium and 29% in helium and 29% in hydrogen. The ultimate tensile strength was 158 and 162 ksi in helium and hydrogen, respectively. Notched strength was 233 and 227 ksi in helium and hydrogen, respectively.

✓ A-286 is a suitable material for use in the NERVA SSBV in the radiation environment of 10^{17} n/cm² (fast) with liquid and gaseous hydrogen in the predicted temperature ^{range} of 40° to 540°R. No additional radiation effects testing is required and no REIRS have been issued.

TABLE 1
SSBV EC-90261

Major Subassemblies (Ref. 2)

Subassembly	Material	
	Prime	Backup
Body	6061-T6	A-286
Bearings, Ball	440-C	
Ball	CRES 347 Electrolized Seal Seat	
Seal	Phosphor Bronze ³ Gold plated Seal Surface	A-286 Gold plated Seal Surface
Ball Support Assy	347	
Seal Lift Assy	347	
Seal Bellows	321	
Seal, Static	321 (lead coated)	
Double Slider Coupling	347	
Bolts	A-286	

5581
EC-90261

AL 6061 - T6

TEST SUMMARY
No. of DATA POINTS

	30-40			100			540		
	None	Low	MED	HIGH	None	Low	MED	HIGH	None
(4) FLOW					4"			4	
(5) FLOW	5"	3"			4"			5	
(6) FLOW	5"	5"	4"		4"		4	5"	

Average Elongation, %

	30-40			100			540		
	None	Low	MED	HIGH	None	Low	MED	HIGH	None
(4) FLOW					7"			5	
(5) FLOW	30"	30"			7"			16	
(6) FLOW	24"	24"	18"		21"		15"	18"	

* Fluence, n/cm² (E > 1 MeV)
LOW: Less than 1.2 x 10¹⁷
MED: 1.2 to 6.0 x 10¹⁷
HIGH: ABOVE 6 x 10¹⁷

FORMS
FACSIMILE (DISAL PLATE & RING)
PLATE (> 0.25" thick)
SHEET (< 0.25" thick)

NOTE: 550V Fast Fluence (10hr)
~ 1.5 x 10¹⁶ n/cm²

Minimum fast fluence 6 x 10¹⁶ n/cm²
(Defc) in test summary data

REFERENCES:

- (a) GTR-19, RH-S-0540, GDFW FER 3421
- (b) REP, Lockheed-Georgia, LAC/ER-8134
- (c) GTR-16, RH-S-02901, RH-S-0123

SSBV
EC 90211

307 children's Skel
(Cont'd)

* Fluence, m/cm^2 (E-1 Skel)

Low: Less than 2×10^{12}

Med: 2×10^{12} to 1×10^{13}

High: Greater than 1×10^{13}

(1) annealed at 500°C

References

- (a) GTR-19, RM-S-0580 / GTR-343-1
- (b) GTR-19, Lockheed-Corning, INC GTR-343-1
- (c) GTR-19, RM-S-0390 / RM-S-0128
- (d) GTR-200, Preliminary Data

1. SSBV Fast Fluence (10hr.) ~ $1.5 \times 10^{16} \text{ m/cm}^2$
minimum fast fluence (data points) $5 \times 10^{16} \text{ m/cm}^2$ (Ref. a)

TABLE 3

EFFECT OF IRRADIATION ON TENSILE PROPERTIES
ALUMINUM 6061-T6

Form	Specification	Fast Fluence n/cm ² (E > 1 MeV)	Irr. Temp. °R	Test Temp °R	Yield Strength KSI		Ultimate Strength KSI		Elongation %		Ref
					Unirr.	Irr.	Unirr.	Irr.	Unirr.	Irr.	
Sheet	QQ-A-327B	3 x 10 ¹⁷	40	40	53	72	74	77	26	18	(4)
Sheet	QQ-A-327B	2.2 x 10 ¹⁷	40	40*	53	63	74	71	26	22	(4)
Sheet	QQ-A-327B	5 x 10 ¹⁶	40	40	47	55	65	69	24	24	(3)
Plate	QQ-A-327B	1 x 10 ^{17**}	30	30	50	59	68	65	30	30	(5)
Sheet	QQ-A-327B	3 x 10 ¹⁷	140	140	50	58	61	62	21	17	(4)
Forging	QQ-A-327B	1.1 x 10 ¹⁸	140	140	52	62	64	66	17	14	(4)
Forging	QQ-A-327B	1.1 x 10 ¹⁸	140	140	54	64	63	66	7	7	(4)
Forging	QQ-A-327B	1.1 x 10 ¹⁸	140	140*	54	61	63	64	7	11	(4)

* - Annealed at ~ 540°R prior to tensile testing

** - E > 0.5 MeV

SSNV
EC 90261

Table 4
347 Stainless Steel

TEST SUMMARY
No. of Data Points

TEST TEMP (°F)
EQUIPMENT
FORM
REFERENCE

	30-40				140				540			
	None	Low	Med	High	None	Low	Med	High	None	Low	Med	High
(1) Round Bar	1	4							4			
(4) PLATE	1		4		4		4		4			
(3) SHEET	1	4							4			
(5) TUBING	4	4							4			
(5) Forging					4		4	7 90				

Average Elongation, %

TEST TEMP (°F)
EQUIPMENT
FORM
REFERENCE

	30-40				140				540			
	None	Low	Med	High	None	Low	Med	High	None	Low	Med	High
(5) Round Bar	41	37							60			
(4) PLATE	27		26		32		33		45			
(3) SHEET	40	41 33							45			
(3) CASTING	8	9							31			
(8) Forging					56		53	52 to 53 50 to 53				

58

TABLE 5
EFFECT OF IRRADIATION ON TENSILE PROPERTIES
347 STAINLESS STEEL

Form	Specifications	Fast Fluence n/cm ² (E > 1 MeV)	Irr. Temp. °R	Test Temp °R	Yield Strength KSI		Ultimate Strength KSI		Elongation %		Ref.
					Unirr.	Irrad.	Unirr.	Irrad.	Unirr.	Irrad.	
Plate	AMS 5512	3 x 10 ¹⁷	40	40	65.6	87.3	230.0	228.4	27.2	26.2	(4)
Welded Plate	AMS 5512	3 x 10 ¹⁷	40	40	67.1	90.0	203.8	208.3	21.9	19.7	(4)
Wrought Sheet	Mil-S-6721B	5 x 10 ¹⁶	40	40	87.4	84.5	259.9	227.1	40.2	40.9	(3)
Wrought Sheet	Mil-S-6721B	5 x 10 ¹⁶	40	40*	87.4	92.4	259.9	236.3	40.2	32.7	(3)
Sand Casting	AMS-5363B	5 x 10 ¹⁶	40	40	93.3	80.3	117.3	115.4	7.7	8.5	(3)
Round Bar	ASTM-A-167-158	1 x 10 ^{17**}	30	30	51.9	64.0	237.4	250.7	41.3	37.0	(5)
Plate	AMS-5512	3 x 10 ¹⁷	140	140	65.4	85.2	213.6	210.4	32.3	33.1	(4)
Welded Plate	AMS-5512	4 x 10 ¹⁷	140	140	54.8	77.6	208.4	204.7	27.1	28.1	(4)
Forging	QQ-S-763C	6 x 10 ¹⁷	140	140	45.4	68.0	176.8	177.2	55.6	52.6	(8)
Forging	QQ-S-763C	1 x 10 ¹⁸	140	140	45.4	86.0	176.8	178.6	55.6	51.9	(8)
Forging	QQ-S-763C	5 x 10 ¹⁸	140	140	45.4	121.9	176.8	179.4	55.6	53.1	(8)
Forging	QQ-S-763C	5 x 10 ¹⁸	140	140*	45.4	106.9	176.8	180.2	55.6	51.9	(8)
Forging	QQ-S-763C	5 x 10 ¹⁸	140	140*	45.4	109.8	176.8	183.1	55.6	32.6	(8)
Forging	QQ-S-763C	5 x 10 ¹⁸	140	140*	45.4	107.3	176.8	179.8	55.6	49.7	(8)

* Annealed at ~540°R prior to tensile testing

** E > 0.5 MeV

TABLE 6
A-286
TEST SUMMARY

558V
EC 9036

No. of DATA POINTS

TEST TEMP (°F) Fluence x Time	30-40				140				540			
	None	Low	Med	High	None	Low	Med	High	None	Low	Med	High
(1) FOS 1/16"	4		4		4		3	5	5			
(2) ROD	4	4			4			4	4			
(3) Bolt	4	4			4		8	3				

Average Elongation, %

TEST TEMP (°F) Fluence x Time	30-40				140				540			
	None	Low	Med	High	None	Low	Med	High	None	Low	Med	High
(1) FOS 1/16"	26		23		52		33	29	24			
(2) ROD	35	31			70			31	26			
(3) Bolt	38	32							24			

* - Fluence, n/m² (E71101)
Low: Less than 2x10¹³
Med: 2x10¹³ to 1x10¹⁴
High: Greater than 1x10¹⁴

FORMS
Form 16, Disc/Plate
Red 0.50" dia.
Plate 0.25"
Bolt, Fluence Closure

(1) Annealed at 590°F
prior to testing at 140

References:

- (a) ER-29, RLS-0520, GOFW FER 342-1
- (b) PRR, Lockheed-Georgia, INC ER-5834
- (c) GTR-14, RLS-0290, RLS-0128
- (d) GTR-200 (Preliminary) data

NOTE: 558V Fast Fluence (10hr.) 1x15x10¹⁶ m²
Minimum fast fluence 5x10¹⁶ m² (RSL-0)
in test summary data

TABLE 7
EFFECT OF IRRADIATION ON TENSILE PROPERTIES
A-286 STAINLESS STEEL

Form	Specification	Fast Fluence n/cm^2 (E 1 MeV)	Irr. Temp. °R	Test Temp °R	Yield Strength KSI		Ultimate Strength KSI		Elongation %		Ref.
					Unirr.	Irrad	Unirr.	Irrad.	Unirr.	Irrad.	
Forging	AMS-5735	3×10^{17}	40	40	146.8	156.5	208.5	204.4	25.8	23.0	(4)
Plate	AMS-5525	5×10^{16}	40	40	135.9	137	223.7	219.8	37.5	32.2	(3)
Fl. Rod	AMS-5735	1×10^{17}	30	30**	149.6	152	235	229	34.5	33.5	(5)
Fl. Rod	AMS-5737	1×10^{17}	30	30**	156.8	165.7	238.2	238	31.8	31.0	(5)
Forging	AMS-5735	3×10^{17}	140	140	131.5	136	197	199.6	31.7	33.4	(4)
Forging	AMS-5735	1.2×10^{18}	140	140	130.6	147	193.9	196.6	28.8	28.7	(4)
Forging	AMS-5735	1.2×10^{18}	140	140	130.6	141.6	193.9	190.0	28.8	31.0	(4)
Bolt	AMS-5735 CW	4.3×10^{17}	140	140	223	223	258	259	16	15***	(9)
Bolt	AMS-5735 CW	1×10^{18}	140	140	223	239	258	259	16	13***	(9)
Bolt	AMS-5735 CW	6.5×10^{18}	140	140	223	247	258	262	16	10***	(9)
Bolt	AMS-5735 CW	6.5×10^{18}	140	140*	223	247	258	256	16	10***	(9)
Bolt	AMS-5735 CW	6.5×10^{18}	140	140*	223	243	258	255	16	11***	(9)

* Annealed at 540° prior to tensile testing

** E 0.5 MeV

*** Bench Elongation

REFERENCES

- (1) Specification No. EC-90261. Detail Specification Part 1, Performance/Design and Qualification Requirements for Structural Support Block Valve and Actuator
- (2) AGC Print No. 1138661 Structural Support Block Valve (SSBV), Ball Concept
- (3) GTR-16 Radiation Effects Test on Structural Materials at -423°F, ANSC Report RN-S-0290
- (4) GTR-19 Test to Determine Cryogenic Radiation Effects on Structural Materials, ANSC Report RN-S-0520
- (5) Effect of Nuclear Radiation on Materials at Cryogenic Temperature, C. A. Schwanbeck, Lockheed Nuclear Products, LAC ER-8434, NASA CR-54881
- (6) Effects of High-Pressure Hydrogen and Metals at Ambient Temperature, W. T. Chandler, et al, Feb. 69, Rocketdyne R-7780-1
- (7) Cryogenic Radiation Effects on NERVA Structural Materials, J. J. Lombardo, C. E. Dixon, and J. A. Begley, et al, ASTM STP 426, 1967
- (8) Preliminary Data GTR-20C, Test No. 37 NRO 027
- (9) Preliminary Data GTR-20C, Test No. 44 NRO 0101A

IDENTIFICATION NO. 631RADIATION EFFECTS INFORMATION REQUESTDATE 21 Dec. 1970SUBJECT 321 Stainless Steel FUNCTIONAL CATEGORY TEST & EVALUATIONREF. ITEM SUPPORT SSBV EC 90261 REF. DATA ITEMS S-110INFORMATION DESIRED Effects of irradiation on tensile strength and ductility
(PURPOSE OF STUDY OR EXPERIMENT)DATE REQUIRED: CY 71RELATION OF ITEM TO ENGINE SSBV Bellows Seal - Station 35.50 & 26.65 at R 25.75
(LOCATION)PER DRAWINGS, DATA ITEMS AGC Print #1138661 - Structural Support Block Valve ConceptENVIRONMENT (4): NEUTRON FLUX N.A. NEUTRON FLUENCE 1×10^{17} n/cm² (E 1 MeV)
(Ref. data items)GAMMA DOSE RATE N.A. GAMMA DOSE N.A.TEMPERATURE 40°R NUCLEAR HEATING RATE N. A.PRESSURE Ambient OTHER N. A.FLUID Hydrogen (Liquid)CYCLE REQUIREMENTS N. A.RELIABILITY ESTIMATE Screening REQUESTED BY W. E. French DATE 12/21/70CONFIDENCE LEVEL Screening APPROVED M. Davidson DATE APPROVED T. Redfield DATE SPECIAL CONSIDERATIONS N. A. APPROVED C. E. Dixon DATE PERTINENT REFERENCES N. A.

IDENTIFICATION NO. 632RADIATION EFFECTS INFORMATION REQUESTDATE 21 Dec. 1970SUBJECT Phosphor Bronze FUNCTIONAL CATEGORY Test & EvaluationREF. ITEM SUPPORT SSBV EC 90261 REF. DATA ITEMS S-110INFORMATION DESIRED Effects of irradiation on tensile strength and ductility as well as fracture toughness
(PURPOSE OF STUDY OR EXPERIMENT)
(if material is brittle)DATE REQUIRED: CY 71RELATION OF ITEM TO ENGINE SSBV Seal - Station 35.50 & 26.65 at R 25.75
(LOCATION)PER DRAWINGS, DATA ITEMS AGC Print #1138661 - Structural Support Block Valve ConceptENVIRONMENT (4): NEUTRON FLUX N.A. NEUTRON FLUENCE 1×10^{17} n/cm² (E 1 MeV)
(Ref. data items)GAMMA DOSE RATE N.A. GAMMA DOSE N. A.TEMPERATURE 40°R NUCLEAR HEATING RATE N.A.PRESSURE Ambient OTHER N.A.FLUID Hydrogen (Liquid)CYCLE REQUIREMENTS N.A.RELIABILITY ESTIMATE Screening REQUESTED BY W. E. French DATE 12/21/70CONFIDENCE LEVEL Screening APPROVED M. Davidson DATE APPROVED T. Redfield DATE SPECIAL CONSIDERATIONS N. A. APPROVED C. E. Dixon DATE PERTINENT REFERENCES N. A.

17
s110-XAV12-W181f1

RADIATION EFFECTS ANALYSIS REPORT, PRESSURE VESSEL AND CLOSURE

REVISION 1

NERVA PROGRAM, CONTRACT SNP-1

SEPTEMBER 1970

REVISED DECEMBER 1970

CONTENTS

Page

1.0	Summary	
2.0	Introduction	
3.0	Environmental Requirements	
3.1	Design Properties	
4.0	Analysis	
4.1	Aluminum 7039-T63 Forging	
4.1.1	Test Experience	
4.1.2	Related Data	
4.1.3	Data Sources Other than NERVA	
4.1.4	Conclusions	
4.2	Aluminum 7075-T73 Forging	
4.2.1	Test Experience	
4.2.2	Related Data	
4.2.3	Data Sources Other than NERVA	
4.2.4	Conclusions	
4.3	Aluminum 6061-T6 Forging	
4.3.1	Test Experience	
4.3.2	Related Data	
4.3.3	Data Sources Other than NERVA	
4.3.4	Conclusions	
4.4	A-286 Stainless Steel Bolts	
4.4.1	Test Experience	
4.4.2	Related Data	
4.4.3	Data Sources Other than NERVA	
4.4.4	Conclusions	
4.5	MP35N Multiphase Bolts	
	References	

Tables

Page

Table I - Canidate Materials - Pressure Vessel and Closure

Figures

Page

Figure 1 - AL 7039-T63 Forging, Test Experience and Operating Conditions for Pressure Vessel

Figure 2 - AL 7075-T73 Forging, Test Experience and Operating Conditions for Pressure Vessel

Figure 3 - AL 6061-T6 Forging, Test Experience and Operating Conditions for Pressure Vessel

Figure 4 - A286 Bolts, Test Experience and Operating Conditions for Pressure Vessel

Radiation Effects Information Requests (REIR)

Page

623 - Aluminum 7039-T63

624 - Aluminum 7075-T73

625 - Aluminum 6061-T6

627 - MP35N Multiphase

s-110 RADIATION EFFECTS ANALYSIS REPORT

PRESSURE VESSEL AND CLOSURE

1.0 SUMMARY

An analysis of the availability of data for the candidate pressure vessel and closure materials with respect to the anticipated nuclear environment and the properties required for design has been completed. The information required to complete the acquisition of suitable data has been outlined on the appropriate Radiation Effects Information Request (REIR) forms. This document is periodically updated in accordance with available information.

Previous reactor tests have supplied information on tensile properties within the general fluence/temperature range of the NERVA requirements for the pressure vessel and closure subsystem except for the MP35N Multiphase nickel-cobalt base alloy. Some of the expected radiation effects on aluminum alloys are inferred from experimental data for similar material conditions or from data related to forms other than forgings. The worst conditions, i.e., low temperature, high fluence, have not been covered in the experimental work. In addition, fracture toughness information is required, as well as tensile properties, due to the low ductility expected in most of the alloys at the extreme environmental conditions.

The REIR's* identifying these problems are (Pages thru):

623 - Aluminum 7039-T63

624 - Aluminum 7075-T73

* Radiation Effects Information Requests (REIR's) document the existence of a need for information at the present time. The solution to the problems noted in the REIR's may be by analysis of existing data, literature, a test program, or a combination of these. The solution is beyond the scope of this document. It should be noted, however, that tests are planned for CY 71 which may supply all or part of the requested information.

625 - Aluminum 6061-T6

626 - A-286 Stainless Steel

627 - MP35N Multiphase

2.0 INTRODUCTION

The pressure vessel and closure subsystem, specification EC-90154,⁽¹⁾ consists primarily of a cylinder and closure, and the bolts for closure. The pressure vessel interfaces with the lower thrust structure and the nozzle. (Drawings 1138077, 1137400).⁽²⁾⁽³⁾

The purpose of this analysis is to determine the availability of suitable data to evaluate the effects of radiation on the pressure vessel and closure subsystem candidate materials and to determine what data may be required to complete a satisfactory analysis.

3.0 ENVIRONMENTAL REQUIREMENTS

The extreme environmental conditions are listed below (10-hour exposure) based on a graphite core:

Nuclear, n/cm ²	>1.0 MeV	7.6 x 10 ¹⁸
n/cm ²	<0.48 eV	1.3 x 10 ¹⁹
Gamma, Rads		3.5 x 10 ¹⁰
Temperature, °R		140 - 760 (Steady State) 40 (Minimum Transient)

3.1 Design Properties

Those properties, required for component design, for the environmental conditions specified are ultimate tensile strength, yield strength, elongation, fracture toughness, and fatigue.

4.0 ANALYSIS

This discussion is concerned with the materials of construction of the pressure vessel cylinder, the closure, and the bolts used at the cylinder/closure interface. The candidate materials for the pressure vessel are listed in Table I.

4.1 Aluminum 7039-T63 Forging

4.1.1 Test Experience

Test data for aluminum 7039-T63 forgings are limited to that recently generated during NERVA Test GTR-20C. Tensile tests were conducted at 140°R after irradiation at the same temperature to 4×10^{17} , 9×10^{17} and 6×10^{18} n/cm² (> 1.0 MeV), and fracture toughness data were obtained at 1×10^{18} and 6×10^{18} n/cm². Some annealing tests were conducted with tensile specimens. The analysis of the above data is incomplete at this date, but preliminary results indicate that elongation decreased from a mean of 12.8% (unirradiated) to 4.4% at 6×10^{18} n/cm² with nearly complete annealing at 540°R. Yield strength increased with irradiation, but did not return to normal after annealing at 540°R. These test variables are illustrated graphically in Figure 1, along with the pressure vessel requirements.

4.1.2 Related Data

Aluminum 7039-T61 and 7039-T64 were tested at elongation temperature (140°R) after irradiation to 1×10^{17} n/cm² > 1.0 MeV in Test GTR-21⁽⁴⁾. The results for the T61 condition might be expected to be similar to the expected results for the T63 condition required in this analysis. Aluminum 7039-T61 plate showed a significant decrease in ductility, due to this low level of irradiation, 16.2% to 14.0% elongation, and the elongation of specimens annealed at 540°R was still significantly lower than

that for controls. Yield strength increased typically due to irradiation, and annealing was only partially effective. Weldments were more severely affected by irradiation than parent metal.

Previous test data have been analyzed as a consolidated group of various aluminum alloys. This analysis concluded that the various aluminum alloys and their heat treatments and forms exhibit considerable variation in tensile properties, but in general, their response to radiation does not differ greatly over the experimental range of temperatures and fluences⁽⁵⁾⁽⁶⁾. Elongation decreased upon irradiation for most materials, including 7039-T61 plate, 7075-T73, and 6061-T6, at 40° to 140°R, but generally showed no change at 540°R. Yield strength of all the materials increased upon irradiation at 40° and 140°, but showed little change at 540°R.

4.1.3 Data Sources Other than NERVA

The data for aluminum alloys obtained from sources other than NERVA include irradiation at cryogenic temperatures to fast fluences below 1×10^{18} n/cm². Investigators report that evaluation of the available data for aluminum radiation effects was difficult due to the variation in the data⁽⁷⁾. Some analysis work has been done by Lockheed Nuclear Products Company, resulting in general categories for probable radiation damage of various alloys⁽⁷⁾⁽⁸⁾. Data for the specific alloys and conditions required for the pressure vessel application are very limited, none being available for 7039-T63.

4.1.4 Conclusions

The ductility of aluminum 7039-T63 forging will decrease to that which is characteristic of brittle materials at 140°R after the anticipated fluence level experienced by the pressure vessel.

The welded material appears to be affected by the low temperature, radiation environment to a greater extent than the parent metal.

Annealing of the aluminum at 540°R does not insure healing of all radiation damage.

Design information for fatigue properties of irradiated material is not available.

REIR No. 623 shows the requirements for required information based on this analysis.

4.2 Aluminum 7075-T73 Forging

4.2.1 Test Experience

The available data for 7075-T73 are tensile test results from six irradiated specimens at 140°R and $9 \times 10^{17} \text{ n/cm}^2 > 1.0 \text{ MeV}$, as illustrated in Figure 2. This test was performed in GTR-19⁽⁹⁾. The ductility of this material showed a small but significant drop due to irradiation to a mean elongation of 12.2% from 13.4%. This compares with aluminum 7039-T63 tests showing the same change, but all values were approximately 2% elongation lower for the latter. At $6 \times 10^{18} \text{ n/cm}^2$, the 7039-T63 elongation dropped to 4.4%, and a similar drop may be expected for the ductility of 7075-T73. GTR-19 tests did not show conclusive healing of radiation damage after annealing at 540°R.

4.2.2 Related Data

Related data, including similar alloys, were discussed in 4.1.2 above.

4.2.3 Data Sources Other than NERVA

(See 4.1.3 above)

4.2.4 Conclusions

The ductility of aluminum 7075-T73 was approximately 2% elongation higher than 7039-T63 and similarly affected by radiation at the lower fluence levels. If the effect is also similar at the maximum required fluence level for the pressure vessel, the 7075-T73 should be classified as a brittle material at 140°R. The effect of annealing on the tensile properties at the higher fluences is uncertain.

REIR No. 624 reflects the need for information at higher fluences, brittle material data, and annealing.

4.3 Aluminum 6061-T6 Forging

4.3.1 Test Experience

NERVA related data for 6061-T6 have been generated for tensile properties in Test GTR-19⁽⁹⁾. The test temperature/fluence combinations used for forging specimens and those anticipated for the pressure vessel are shown graphically in Figure 3. Forgings irradiated to 1×10^{18} n/cm² (> 1.0 MeV) in GTR-19 dropped in elongation from 16% to 14% due to irradiation when tested at 140°R, and welded forgings dropped from 19% to 8% elongation.

Aluminum 6061-T6 forging material appears to be similar to 7039-T63 and 7075-T73 after exposure to 1×10^{18} n/cm² at 140°R (see 4.1 and 4.2 above). Thus, aluminum 6061-T6 may be expected to show greatly reduced ductility at 140°R after exposure to radiation fluences expected for the pressure vessel.

4.3.2 Related Data

Tensile test data for 30 parent metal specimens and 11 welded specimens plus unirradiated controls are available from Tests GTR-16⁽¹⁰⁾ and GTR-19⁽⁹⁾. Fluences varied from 6×10^{16} to 1×10^{18} n/cm² > 1.0 MeV, and some

specimens were annealed prior to testing. Tests were conducted at 40°R and 140°R. Material forms tested were disc and ring forgings, plate and sheet.

Plate material, irradiated and tested at 37°R in GTR-16 indicated no change in ductility due to irradiation to $5 \times 10^{16} \text{ n/cm}^2 > 1.0 \text{ MeV}$. Plate material, irradiated to $3 \times 10^{17} \text{ n/cm}^2 > 1.0 \text{ MeV}$ in GTR-19, exhibited a significant drop in ductility due to irradiation both at 140°R and 40°R (21% to 16%, and 26% to 18%, respectively). The radiation damage appeared to be greater at 40°R than at 140°R with respect to both yield strength and elongation.

4.3.3 Data Sources Other than NERVA

(See 4.1.3 above)

4.3.4 Conclusions

Based on the effects of test temperature and radiation at the low fluence levels of the previous aluminum 6061-T6 tests, this material may be expected to be brittle in the pressure vessel environment as inferred from the results of other alloys.

The welded forgings were more severely damaged by irradiation than were the parent metal forgings.

No fatigue data are available for irradiated material.

REIR No. 625 reflects the requirement for tensile and fracture toughness data at higher fluences for parent and welded forging material. The required fatigue data is also noted.

4.4 A-286 Stainless Steel Bolts

4.4.1 Test Experience

A-286 stainless steel forging and bolt radiation experience is illustrated in Figure 4. The pressure vessel requirements for temperature and fluence are also shown.

The bolt specimens tested in GTR-20C were irradiated and tested at 140°R after exposure to fluences of 4×10^{17} , 1×10^{18} , and 6×10^{18} n/cm² (> 1.0 MeV). Notched and fatigue cracked bolts were also tested (analysis not complete at this date). Preliminary results of tests on unnotched bolts indicate that the elongation decreased from a mean of 15.8% to 10.2% at the highest fluence, while the .2% offset yield strength increased from a mean of 223 to 243 ksi and ultimate strength remained constant. Annealing the high fluence-exposed specimens resulted in no appreciable recovery of elongation or yield strength values.

4.4.2 Related Data

The A-286 test experience also includes forging, rod, and plate specimen tensile tests from GTR-16⁽¹⁰⁾, GTR-19⁽⁹⁾, and from Lockheed-Georgia screening tests at Plum Brook⁽⁸⁾. These data include 23 irradiated specimens at 1×10^{17} to 1×10^{18} n/cm² > 1.0 MeV, tested at 40°R and 140°R. Four specimens were annealed prior to testing.

The A-286 plate specimens, irradiated to 5×10^{16} n/cm² (> 1.0 MeV) at 37°R and tested at 37°R in Test GTR-16, showed a mean decrease of 13% in ductility (elongation) due to the irradiation. A-286 forging specimens were irradiated to 3×10^{17} and 1×10^{18} n/cm² (> 1.0 MeV) in GTR-19 with no effect on ductility at 140°R. However, when A-286 forging specimens were irradiated to 3×10^{17} n/cm² (> 1.0 MeV) at 40°R and tested at 40°R a small but significant decrease (from 26% to 23% elongation) in ductility was noted.

These data are not necessarily representative for A-286 bolts which are cold worked and aged. GTR-20C included irradiation of bolts as described in 4.4.1.

4.4.3 Data Sources Other Than NERVA

Irradiated A-286 steel has been tested at room temperature and above. No data have been obtained at cryogenic temperatures, outside of NERVA related work.

4.4.4 Conclusions

REIR No. 626 covers a request for information on tensile, fracture toughness, and fatigue properties of irradiated bolts. A portion of this request may be satisfied by the analysis and reporting of GTR-20C data.

4.5 MP35N Multiphase Bolts

The MP35N Multiphase, nickel-cobalt base alloy, has not been tested for radiation effects. REIR No. 627 requests information on the effects of fluence and temperature, in the anticipated pressure vessel environment, on the tensile properties, fracture toughness, and fatigue strength.

REFERENCES

1. EC-90154, Detail Specification, Part 1, Performance/Design and Qualification Requirements for Pressure Vessel and Closure System.
2. Drawing No. 1138077, Layout, NERVA Pressure Vessel and Closure.
3. Drawing No. 1137400, 75K NERVA Flight Engine Layout, Full Flow.
4. RN-S-0553, Final Report, Cryogenic Radiation Effects on Aluminum 7039 and Hastelloy X, GTR-21 Test 37R018, Aerojet Nuclear Systems Co., April 1970.
5. Consolidation and Analysis of Radiation Effects Data (CARD), Data Bank.
6. s110-TAA11-W181f7d Multiple Regression Analysis of Consolidated Radiation Effects Data, NERVA Materials, Tensile Properties of Aluminum, September 1970.
7. REIC Report No. 45, The Effects of Neutron Radiation on Structural Materials, Battelle Memorial Institute, 30 June 1967.
8. LAC ER-8434, Effect of Nuclear Radiation on Materials Properties at Cryogenic Temperatures, Lockheed Nuclear Products, January 1965.
9. RN-S-0520, Final Report, GTR-19 Test to Determine Cryogenic Radiation Effects on Structural Materials, Aerojet-General Corp., September 1969.
10. RN-S-0290, Final Report, GTR-16 Radiation Effects Test on Structural Materials at -423°F, Aerojet-General Corp., November 1966.

TABLE I

CANDIDATE MATERIALS - PRESSURE VESSEL AND CLOSURE

<u>Component</u>	<u>Material</u>	<u>Form</u>	<u>Status</u>
Cylinder	Al 7075-T73	Forging	Prime
Cylinder	Al 7039-T63	Forging	Backup
Closure	Al 6061-T6	Forging	Prime
Closure	Al 7039-T63	Forging	Backup
Bolts	MP35N Multiphase		Prime
Bolts	A-286		Backup

FIGURE 1 - AI 7039-T63 FORGING, TEST EXPERIENCE AND OPERATING CONDITIONS FOR PRESSURE VESSEL

FIGURE 2 - AI 7075-T73 FORGING, TEST EXPERIENCE AND OPERATING CONDITIONS FOR PRESSURE VESSEL

FIGURE 3 - Al 6061-T6 FORGING, TEST EXPERIENCE AND OPERATING CONDITIONS FOR PRESSURE VESSEL

FIGURE 4 - A286 BOLTS, TEST EXPERIENCE AND OPERATING CONDITIONS FOR PRESSURE VESSEL

IDENTIFICATION NO. 623

RADIATION EFFECTS INFORMATION REQUEST

DATE 18 September 1970

SUBJECT Aluminum 7039-T63 FUNCTIONAL CATEGORY _____

REF. ITEM SUPPORT Pressure Vessel and Closure Subsystem REF. DATA ITEMS C-002, S-110
EC-90154

INFORMATION DESIRED Determine effects of fluence and temperatures, over the extremes of
(PURPOSE OF STUDY OR EXPERIMENT) the ranges shown below, on tensile properties and fracture toughness and fatigue for parent metal and
welds, including the annealing of radiation damage.

DATE REQUIRED: CY 1971

RELATION OF ITEM TO ENGINE PV interfaces with LTS and nozzle
(LOCATION)

PER DRAWINGS, DATA ITEMS 1138077, 1137400

ENVIRONMENT : NEUTRON FLUX _____ NEUTRON FLUENCE 7.6×10^{18} n/cm² >1.0 MeV
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE 3.5×10^{10} Rads

TEMPERATURE 40 - 760°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER 1.3×10^{19} n/cm² < 0.48 eV

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99% REQUESTED BY D. A. Brown DATE 18 Sept. 1970

CONFIDENCE LEVEL .95 APPROVED H. L. Springer DATE _____

APPROVED T. A. Redfield DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED C. E. Dixon DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 624

RADIATION EFFECTS INFORMATION REQUEST

DATE 18 September 1970

SUBJECT Aluminum 7075-T73 FUNCTIONAL CATEGORY _____

REF. ITEM SUPPORT Pressure Vessel and Closure Subsystem REF. DATA ITEMS C-002, S-110
EC-90154

INFORMATION DESIRED Determine effects of fluence and temperature, over the extremes of the
(PURPOSE OF STUDY OR EXPERIMENT) ranges shown below, on the tensile properties and fracture toughness, including the annealing of radiation
damage.

DATE REQUIRED: CY 1971

RELATION OF ITEM TO ENGINE PV interfaces with LTS and nozzle
(LOCATION)
PER DRAWINGS, DATA ITEMS 1138077, 1137400

ENVIRONMENT : NEUTRON FLUX _____ NEUTRON FLUENCE 7.6×10^{18} n/cm² > 1.0 MeV
(Ref. data items) GAMMA DOSE RATE _____ GAMMA DOSE 3.5×10^{10} Rads
TEMPERATURE 40 - 760°R NUCLEAR HEATING RATE _____
PRESSURE _____ OTHER 1.3×10^{19} n/cm² < 0.48 eV
FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99% REQUESTED BY D. A. Brown DATE 18 Sept. 1970

CONFIDENCE LEVEL .95 APPROVED H. L. Springer DATE _____

APPROVED T. A. Redfield DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED C. E. Dixon DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 625

RADIATION EFFECTS INFORMATION REQUEST

DATE 18 September 1970

SUBJECT Aluminum 6061-T6 FUNCTIONAL CATEGORY _____

REF. ITEM SUPPORT Pressure Vessel and Closure Subsystem REF. DATA ITEMS EC-90154, S-110

INFORMATION DESIRED Determine effects of fluence and temperature, over the ranges of the
(PURPOSE OF STUDY OR EXPERIMENT) ranges shown below, on tensile properties and fracture toughness and fatigue for parent metal and welds.

DATE REQUIRED: CY 1971

RELATION OF ITEM TO ENGINE PV interfaces with LTS and nozzle
(LOCATION)

PER DRAWINGS, DATA ITEMS 1138077, 1137400

ENVIRONMENT : NEUTRON FLUX _____ NEUTRON FLUENCE 7.6×10^{18} n/cm² > 1.0 MeV
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE 3.5×10^{10} Rads

TEMPERATURE 40 - 760°C NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER 1.3×10^{19} n/cm² < 0.48 eV

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99% REQUESTED BY D. A. Brown DATE 18 Sept. 1970

CONFIDENCE LEVEL .95 APPROVED H. L. Springer DATE _____

APPROVED T. A. Redfield DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED C. E. Dixon DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 626

RADIATION EFFECTS INFORMATION REQUEST

DATE 18 September 1970

SUBJECT A-286 Stainless Steel FUNCTIONAL CATEGORY

REF. ITEM SUPPORT Pressure Vessel and Closure Subsystem REF. DATA ITEMS C-002, S-110

EC-90154

INFORMATION DESIRED Determine effects of fluence and temperature, over the extremes of
(PURPOSE OF STUDY OR EXPERIMENT) the ranges shown below, on tensile properties and fracture toughness and fatigue.

DATE REQUIRED: CY 1971

RELATION OF ITEM TO ENGINE PV interfaces with LTS and nozzle
(LOCATION)

PER DRAWINGS, DATA ITEMS 1138077, 1137400

ENVIRONMENT : NEUTRON FLUX NEUTRON FLUENCE 7.6×10^{18} n/cm² > 1.0 MeV
(Ref. data items)

GAMMA DOSE RATE GAMMA DOSE 3.5×10^{10} Rads

TEMPERATURE 40 - 760°R NUCLEAR HEATING RATE

PRESSURE OTHER 1.3×10^{19} n/cm² < 0.48 eV

FLUID

CYCLE REQUIREMENTS

RELIABILITY ESTIMATE 99% REQUESTED BY D. A. Brown DATE 18 Sept. 1970

CONFIDENCE LEVEL .95 APPROVED H. L. Springer DATE

APPROVED T. A. Redfield DATE

SPECIAL CONSIDERATIONS APPROVED C. E. Dixon DATE

PERTINENT REFERENCES

IDENTIFICATION NO. 627

RADIATION EFFECTS INFORMATION REQUEST

DATE 18 September 1970

SUBJECT MP35N Multiphase FUNCTIONAL CATEGORY

REF. ITEM SUPPORT Pressure Vessel and Closure Subsystem REF. DATA ITEMS C-002, S-110
EC-90154

INFORMATION DESIRED Determine effects of fluence and temperature, over the extremes of the
(PURPOSE OF STUDY OR EXPERIMENT) ranges shown below, on tensile properties and fracture toughness and fatigue.

DATE REQUIRED: CY 1971

RELATION OF ITEM TO ENGINE PV interfaces with LTS and nozzle
(LOCATION)
PER DRAWINGS, DATA ITEMS 1388077, 1137400

ENVIRONMENT : NEUTRON FLUX NEUTRON FLUENCE 7.6×10^{18} n/cm² > 1.0 MeV
(Ref. data items) GAMMA DOSE RATE GAMMA DOSE 3.5×10^{10} Rads
TEMPERATURE NUCLEAR HEATING RATE
PRESSURE OTHER 1.3×10^{19} n/cm² < 0.48 eV
FLUID

CYCLE REQUIREMENTS

RELIABILITY ESTIMATE 99% REQUESTED BY D. A. Brown DATE 18 Sept. 1970

CONFIDENCE LEVEL .95 APPROVED H. L. Springer DATE

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SPECIAL CONSIDERATIONS APPROVED C. E. Dixon DATE

PERTINENT REFERENCES

s110-XAN08-W181F1

s110 RADIATION EFFECTS ANALYSIS REPORT-NOZZLE

REVISION 1

NERVA Program

Contract SNP-1

September 1970

REVISED DECEMBER 1970

I. L. Odgers, Manager
Engineering Department
Aerojet Nuclear Systems Company

Classification Category	
UNCLASSIFIED
Classifying Officer	<u>24 Sept 70</u> Date _u

CONTENTS

PAGE

1.0	Summary	
2.0	Introduction	
3.0	Environmental Requirements	
3.1	Design Properties	
4.0	Analysis	
4.1	AISI 347 Forging	
4.1.1	Test Experience	
4.1.2	Related Data	
4.1.3	Data Sources Other than NERVA	
4.1.4	Conclusions	
4.2	AISI 347 Sheet	
4.2.1	Test Experience	
4.2.2	Related Data	
4.2.3	Data Sources Other than NERVA	
4.2.4	Conclusions	
4.3	Hastelloy X Forging	
4.3.1	Test Experience	
4.3.2	Related Data	
4.3.3	Data Sources Other than NERVA	
4.3.4	Conclusions	
4.4	Hastelloy X Sheet	
4.4.1	Test Experience	
4.4.2	Related Data	
4.4.3	Data Sources Other than NERVA	
4.4.4	Conclusions	

Contents (Continued)

- 4.5 ARMC0 22-13-5 Forging
- 4.5.1 Test Experience
- 4.5.2 Related Data
- 4.5.3 Data Sources Other than NERVA
- 4.5.4 Conclusions

Tables

Page

Table I - Candidate Materials - Nozzle

Table 2 - Summary of Results - AISI 347 Sheet and
Plate Radiation Tests

Figures

Page

Figure 1 - AISI 347 Forging, Test Conditions and
Operating Conditions for Nozzle Jacket

Figure 2 - AISI 347 Sheet, Test Conditions and Operating
Conditions for Nozzle Tubes

Figure 3 - Hastelloy-X Forging, Test Conditions and
Operating Conditions for Nozzle Jacket

Figure 4 - Effect of Radiation on Elongation of Hastelloy-X
Forging Material at 140°F

Figure 5 - Effect of Radiation on Yield Strength of
Hastelloy-X Forging Material at 140°R

Figure 6 - Hastelloy-X Sheet, Test Conditions and
Operating Conditions for Nozzle Tubes

Figure 7 - Effect of Temperature and Radiation on
Elongation of Hastelloy-X Sheet Material
Over Experimental Range

Figure 8 - Effect of Fluence on Elongation of Hastelloy-X
Sheet Material

Radiation Effects Information Requests (REIR)

Page

618 - AISI 347 Forging

619 - AISI 347 Sheet

620 - Hastelloy-X Forging

621 - Hastelloy-X Sheet

622 - ARMCO 22-13-5

S-110 RADIATION EFFECTS ANALYSIS REPORT

NOZZLE

1.0 SUMMARY

An analysis of the availability of data for the candidate nozzle materials with respect to the anticipated nuclear environment and the properties required for design has been completed. The information required to complete the acquisition of suitable data has been outlined on the appropriate Radiation Effects Information Request (REIR) forms. This document is periodically updated in accordance with available information.

Previous reactor tests have supplied information on tensile properties within the general fluence/temperature range of the NERVA requirements for the nozzle candidate materials except for ARMCO 22-13-5. In addition, a review of the data generated for similar materials and forms under applicable conditions, and the available data from sources other than NERVA, has provided further insight for predicted radiation effects.

Fatigue properties at elevated temperatures are required for nozzle tube sheet material. This property has not been evaluated for the irradiated materials. Non-irradiation fatigue tests in progress will determine whether a sufficient margin in fatigue strength exists to preclude testing irradiated tube materials.

The REIR's* identifying these ~~problems~~ ^{problems} are (Pages ~~5 thru~~ ^{6 thru});

- 618 - AISI 347 Forging
- 619 - AISI 347 Sheet
- 620 - Hastelloy-X Forging
- 621 - Hastelloy-X Sheet
- 622 - ARMCO 22-13-5 Forging

*Radiation Effects Information Requests (REIR's) document the existence of a need for information at the present time. ~~The solution to the problems~~ ^{The solution} noted in the REIR's may be by analysis of existing data, literature, a test program, or a combination of these. The solution is beyond the scope of this document. It should be noted, however, that tests are planned for ~~CY 1971~~ ¹⁹⁷² which may supply all or part of the requested information.

2.0 INTRODUCTION

The nozzle, specification DS-90196⁽¹⁾, is a part of the Nozzle Assembly Subsystem (EC-90151)⁽²⁾. It consists primarily of a jacket and tubes, in a brazed assembly. The nozzle interfaces with the pressure vessel.⁽³⁾⁽⁴⁾

The purpose of this analysis is to determine the availability of suitable data to evaluate the effects of radiation on the nozzle candidate materials and to determine what data may be required to complete a satisfactory analysis.

3.0 ENVIRONMENTAL REQUIREMENTS

The extreme environmental conditions are listed below (10-hour exposure) based on a graphite core:

<u>Environment</u>	<u>Nozzle Jacket</u>	<u>Nozzle Tubes</u>
Nuclear, $n/cm^2 > 1.0 \text{ MeV}$	2.4×10^{19}	2.4×10^{19}
$n/cm^2 < 0.48 \text{ eV}$	1.9×10^{18}	1.9×10^{18}
Gamma, Rads	1.5×10^{10}	1.5×10^{10}
Temperature, °R	40-800	40-2060

3.1 Design Properties

Those properties, required for component design, for the environmental conditions specified are as shown below:

<u>Component</u>	<u>Property Values Required</u>			
	<u>Ult. T.S.</u>	<u>Yield T.S.</u>	<u>Elongation</u>	<u>Fatigue</u>
Jacket	X	X	X	X
Tubes	X	X	X	X

4.0 ANALYSIS

This discussion is concerned with the materials of construction of the nozzle. The candidate materials are listed in Table I.

4.1 AISI 347 Forging

4.1.1 Test Experience

Twenty-four AISI 347 forging tensile test specimens have been tested as part of test GTR-20C. The environmental conditions included 5×10^{17} , 1×10^{18} and 5×10^{18} n/cm² (> 1.0 MeV), with irradiation and test temperatures of 140°R. In addition, the effect of annealing at 540°R on tensile properties was evaluated.

These data points are shown graphically in Figure 1, in relation to the environmental requirements of the nozzle jacket.

The analysis of the data is incomplete at this date, but preliminary results indicate that radiation had little, if any, effect on elongation at 140°R, with (chart) values between 46 and 59%. The .2% offset yield strength increased from a mean of 45 ksi to 122 ksi at the highest fluence level, and all values of annealed specimens were in excess of 100 ksi, indicating only partial healing of radiation damage.

4.1.2 Related Data

Additional data which may be related to AISI 347 forgings are available for sheet materials as described below under AISI 347 sheet, Test Experience.

4.1.3 Data Sources Other than NERVA

^{4,2,3}
(See below, AISI 347 Sheet, Data Sources Other than NERVA)

4.1.4 Conclusions

Ductility of AISI 347 forgings does not appear to be affected by radiation at cryogenic temperatures with fluences up to $5 \times 10^{18} \text{ n/cm}^2$ ($> 1.0 \text{ MeV}$). Further information on tensile properties is needed to confirm suitable damage threshold level at anticipated nozzle jacket fluence requirements. REIR No. 618 requests information for fluence to $2.4 \times 10^{19} \text{ n/cm}^2$ ($> 1.0 \text{ MeV}$).

No requirement for radiation effects data at the elevated temperatures exists, since it is assumed that start-of-life design would be applicable.

4.2 AISI 347 Sheet

4.2.1 Test Experience

Test experience with AISI 347 sheet was obtained from

- a) GTR-19⁽⁵⁾, tensile testing at 37°R and 140°R after irradiation to approximately $3 \times 10^{17} \text{ n/cm}^2$ ($> 1.0 \text{ MeV}$), at the same temperatures, welded and non-welded specimens, 0.125 inch thick,
- b) GTR-16⁽⁶⁾, tensile testing at 37°R after irradiation to $5 \times 10^{16} \text{ n/cm}^2$ ($> 1.0 \text{ MeV}$), at approximately 100°F, welded and non-welded specimens, greater than 0.02 inch thick,
- c) GTR-18⁽⁷⁾, tensile testing at elevated temperatures (1500 to 2000°R) after irradiation to $1 \times 10^{19} \text{ n/cm}^2$ ($< 0.48 \text{ eV}$), at approximately 560°R, and
- d) the First Irradiation Test of Structural Materials⁽⁸⁾. These data points are shown graphically in Figure 2, in relation to the environmental requirements of the nozzle tubes. The results of all previous tensile tests on AISI 347 steel are stored in the CARD data bank⁽⁹⁾. An analysis of the consolidated data has not been conducted.

The individual analyses, reported in the referenced reports, resulted in the conclusions summarized in Table II. The general trend appears to be no change in ductility and an increase in yield strength at cryogenic temperatures due to radiation, (to a fluence level of 3×10^{17} n/cm⁽²⁾ (> 1.0 MeV) and a decrease in elongation at high temperatures (33.7% elongation at 2000°R, 1×10^{19} n/cm² (< 0.48 eV)).

4.2.2 Related Data

(See AISI 347 Forging, Test Experience)

4.2.3 Data Sources Other Than NERVA

The data obtained from sources other than the NERVA program indicate that AISI 347 shows an increase in tensile strength and a decrease in ductility at low temperatures (50°C) due to irradiation, while the decrease in ductility at temperatures above 1400°R was described as a "considerable" reduction. AISI 347, irradiated to 1.0×10^{20} n/cm² (fast), showed a loss in total elongation from about 35% at 1400°R to about 15% at 1750°R. AISI 347 tubing indicated a more severe ductility loss than plate material⁽¹⁰⁾.

4.2.4 Conclusions

AISI 347 sheet will exhibit a loss of ductility at elevated temperatures. Although the available data from various sources are not in complete agreement, it is believed that the large ductility losses will not result in reclassifying the material as brittle, since start-of-life ductility is high.

The non-NERVA data suggests that tubing may be more severely damaged by radiation than plate material. Radiation effects on weldments were the same as for parent metal.

REIR No. 619 reflects the need for fatigue data for irradiated material.

No data has been requested for a high fluence, cryogenic temperature environment, since it is not a possible operating condition.

4.3 Hastelloy X Forging

4.3.1 Test Experience

The Hastelloy X forging test data were obtained from a) GTR-19⁽⁵⁾, tensile testing at 37 and 140°R after irradiation to a fluence of 1×10^{18} n/cm² (> 1.0 MeV) at the same temperatures; b) GTR-21⁽¹¹⁾, tensile testing at 140°R after irradiation to 8×10^{16} to 1.4×10^{17} n/cm² (> 1.0 MeV) at the same temperature; and c) GTR-20C, tensile testing of a low aluminum heat at 140°R after irradiation at 140°R to 5×10^{18} n/cm² (> 1.0 MeV), with some specimens annealed at 540°R.

These test conditions are shown graphically in Figure 3, in relation to the environmental exposure requirements of the nozzle jacket.

Data analysis for Test GTR-20C is not complete at this date. Data from the other reactor experiments have been collected and stored in the CARD data bank⁽⁹⁾, and an analysis has been completed on the consolidated data⁽¹²⁾. The data show that elongation decreases and yield strength increases for Hastelloy X forging material irradiated and tested at 140°R. The expected mean values for elongation and yield strength are illustrated in Figures 4 and 5 respectively. The elongation at 1×10^{18} n/cm² (> 1.0 MeV) is predicted to average 13.6%⁽¹²⁾. The material has not been tested at the maximum anticipated fluence of 2.4×10^{19} n/cm², at which low values of elongation would be expected. The preliminary data for the low aluminum heat of Hastelloy X, in Test GTR-20C, show a mean elongation of 14.3% at 5×10^{18} n/cm², at 140°R. Annealing at 540°R increased the elongation only slightly (to 15% - 18%).

Welded material exhibited similar behavior to that of the parent material in Test GTR-19⁽⁵⁾.

The properties of irradiated Hastelloy X forging at temperatures approaching the maximum required for the jacket (800°R) are assumed to contain no design constraints.

4.3.2 Related Data

Other forms of Hastelloy X and other experimental data are not applicable to the conditions of interest for the nozzle jacket

4.3.3 Data Sources Other Than NERVA

Data from other sources within the required environmental range, show increases in tensile strength and decreases in ductility due to irradiation at 50°C⁽¹⁰⁾.

4.3.4 Conclusions

The ductility of Hastelloy X forging material, parent and welded, approaches the brittle stage (10% elongation) at fluence levels above 1×10^{18} n/cm² (> 1.0 MeV) at 140°R. Elongation values at the maximum anticipated nozzle fluence are unknown.

Increasing yield strength with increasing fluence imposes no design constraints.

Effect of radiation at higher temperatures (to 800°R) are of no design interest.

REIR No. 620 covers a request for further information at the maximum fluence anticipated at cryogenic temperature.

4.4 Hastelloy X Sheet

4.4.1 Test Experience

The available data from irradiated materials were obtained from tensile tests in GTR-18 at elevated temperatures, 1500 to 2000°R, after irradiation to 6×10^{17} to 1×10^{19} n/cm² (< 0.48 eV) at approximately 560°R.

These test conditions are shown graphically in Figure 6, in relation to the environmental exposure requirements of the nozzle tubes.

These tensile property data are stored in the CARD data bank⁽⁹⁾ and an analysis has been completed⁽¹²⁾. The constant elongation contour lines covering the temperature-fluence test range are shown in Figure 7. A maximum predicted mean elongation of approximately 33% is indicated for the lower temperatures, lower fluence portions of the test range. The mean predicted elongation decreases to approximately 20% at the highest temperature, highest fluence test condition. These data, along with unirradiated specimen data are replotted in Figure 8. The unirradiated material exhibits an increasing elongation with increasing temperature, while the irradiated material elongation tends to decrease with increasing temperature.

Yield strength was not affected by the radiation environment in the elevated temperature test range.

No radiation effects data for Hastelloy X sheet material have been requested for the cryogenic temperatures, *since it is not applicable to actual operating conditions* ✓

4.4.2 Related Data

None

4.4.3 Data Sources Other Than NERVA

Hastelloy X radiation effects data have been reported for irradiation temperatures of 300°C (1000°R) to 700°C (1750°R) at fluences of 3×10^{19} to 2×10^{21} n/cm² (fast)⁽¹⁰⁾. The decrease in ductility of Hastelloy X

at temperatures above 1400°R was described as "drastic". The elongation decreased to values between 5 and 7.4% at 1700°R after irradiation to fluences of 9×10^{19} to 2×10^{21} n/cm² (fast).

4.4.4 Conclusions

Temperature and fluence interact to produce very large changes in ductility for certain combinations of these test variables. However, at the "worst case" condition predicted for nozzle tubes, 2060°R and 1.9×10^{18} n/cm² (< 0.48 eV), the mean elongation will be approximately 27% (Figure 7), with a lower limit of 17% (Reference 12). This is well above the brittle condition as currently defined for NERVA materials.

No fatigue test data are available for this material in the irradiated condition.

The radiation effects on Hastelloy X sheet at cryogenic temperatures ^{should} ~~would~~ not impose design constraints on nozzle tubes.

REIR No. 621 is a request for information on fatigue properties at the "worst case" conditions.

4.5 ARMCO 22-13-5 Forging

4.5.1 Test Experience

None

4.5.2 Related Data

None

4.5.3 Data Sources Other Than NERVA

None

4.5.4 Conclusions

No data have been generated for the radiation effects on ARMCO 22-13-5 stainless steel forging. REIR No. 621 is a request for information at cryogenic temperatures.

REFERENCES

1. DS-90196, Detail Specification, Part 1, Performance/Design and Qualification Requirements for Nozzle.
2. EC-90151, Detail Specification, Part 1, Performance/Design and Qualification Requirements for Nozzle Assembly Subsystem.
3. Dwg. No. 1138000, Nozzle Layout.
4. Dwg. No. 1138200, Interface Layout
5. RN-S-0520, Final Report, GTR-19 Test to Determine Cryogenic Radiation Effects on Structural Materials, Aerojet-General Corp., September 1969.
6. RN-S-0290, Final Report, GTR-16 Radiation Effects Test on Structural Materials at -423°F, Aerojet-General Corp., Nov. 1966.
7. RN-S-0398, Final Report on GTR-18 Test to Determine High-Temperature Radiation Effects on Structural Materials, Aerojet-General Corp., July 1967.
8. REON 2473, Data Obtained from First Irradiation Test of Structural Materials, Aerojet-General Corp., April 1963.
9. Consolidation and Analysis of Radiation Effects Data (CARD), data bank.
10. REIC Report No. 45, The Effects of Neutron Radiation on Structural Materials, Battelle Memorial Institute, 30 June 1967.
11. RN-S-0553, Final Report, Cryogenic Radiation Effects on Aluminum 7039 and Hastelloy X, GTR-21 Test 37R018, Aerojet Nuclear Systems Co., April 1970.
12. S-110-TAO4-W181-7e, Multiple Regression Analysis of Consolidated Radiation Effects Data, NERVA Materials; Tensile Properties of Hastelloy X (Parent Metal); Aerojet Nuclear Systems Company, August 1970.

TABLE I
CANDIDATE MATERIALS - NOZZLE

<u>Component</u>	<u>Material</u>	<u>Form</u>	<u>Status</u>
Jacket	ARMCO 22-13-5	Forging	Prime
Jacket	Hastelloy-X	Forging	Backup
Jacket	AISI 347	Forging	Backup
Tubes	AISI 347	Sheet	Prime
Tubes	Hastelloy-X	Sheet	Backup

TABLE II

SUMMARY OF RESULTS - AISI 347 SHEET AND PLATE RADIATION TESTS

<u>Form</u>	<u>Parent or Weld</u>	<u>Fluence n/cm² > 1.0 MeV</u>	<u>Irradiation and Test Temp., °R</u>	<u>Ductility Change</u>	<u>Yield Strength Change</u>	<u>Test</u>	<u>Reference</u>
0.25" Plate	Parent	2×10^{17}	40*	Decrease	Increase	First	8
0.25" Plate	Weld	2×10^{17}	40*	None	Increase	First	8
0.125" Sheet	Parent	5×10^{16}	40	None	Decrease	GTR-16	6
0.125" Sheet	Weld	5×10^{16}	40	None	Decrease	GTR-16	6
0.25" Plate	Parent	3×10^{17}	40	None	Increase	GTR-19	5
0.25" Plate	Parent	3×10^{17}	140	None	Increase	GTR-19	5
0.25" Plate	Weld	3×10^{17}	40	None	Increase	GTR-19	5
0.25" Plate	Weld	3×10^{17}	140	None	Increase	GTR-19	5
Thin Sheet	Parent	$1 \times 10^{19**}$	1500-2000	Decrease	None	GTR-18	7

* 30 days at room temperature between irradiation and test.

** Thermal fluence <0.48 eV, max. test.

FIGURE 1 - AISI 347 FORGING, TEST CONDITIONS AND OPERATING CONDITIONS FOR NOZZLE JACKET

FIGURE 2 - AISI 347 SHEET, TEST CONDITIONS AND OPERATING CONDITIONS FOR NOZZLE TUBES

FIGURE 3 - HASTELLOY -X FORGING, TEST CONDITIONS AND OPERATING CONDITIONS FOR NOZZLE JACKET

FIGURE 4 - EFFECT OF RADIATION ON ELONGATION OF
HASTELLOY-X FORGING MATERIAL AT
140°R (REF 12)

FIGURE 5 - EFFECT OF RADIATION ON YIELD STRENGTH OF
HASTELLOY-X FORGING MATERIAL AT 140°R
(REF 12)

FIGURE 6- HASTELLOY - X SHEET, TEST CONDITIONS AND OPERATING CONDITIONS FOR NOZZLE TUBES

FIGURE 7 - EFFECT OF TEMPERATURE AND RADIATION ON ELONGATION OF HASTELLOY-X SHEET MATERIAL OVER EXPERIMENTAL RANGE (REF 12)

FIGURE 8 - EFFECT OF FLUENCE ON ELONGATION OF HASTELLOY-X SHEET MATERIAL (REF 12)

IDENTIFICATION NO. 618

RADIATION EFFECTS INFORMATION REQUEST

DATE 10 September 1970

SUBJECT AISI 347 STEEL FORGING FUNCTIONAL CATEGORY

REF. ITEM SUPPORT NOZZLE, DS-90196 REF. DATA ITEMS C-002, S-110

INFORMATION DESIRED EFFECTS OF FLUENCE AND TEMPERATURE, AT "WORST CASES" IN THE RANGES SHOWN BELOW,
(PURPOSE OF STUDY OR EXPERIMENT)
ON TENSILE PROPERTIES. EVALUATE ANNEALING EFFECTS. INFORMATION SHOULD SUPPLEMENT EXISTING DATA.

DATE REQUIRED: CY 1971

RELATION OF ITEM TO ENGINE COMPONENT OF NOZZLE ASSEMBLY SUBSYSTEM, EC-90151
(LOCATION)
PER DRAWINGS, DATA ITEMS 1138200, 1138000

ENVIRONMENT : NEUTRON FLUX NEUTRON FLUENCE 2.4×10^{19} n/cm² > 1.0 MeV
(Ref. data items) GAMMA DOSE RATE GAMMA DOSE 1.5×10^{10} RADS
TEMPERATURE 40-800°R NUCLEAR HEATING RATE
PRESSURE OTHER 1.9×10^{18} n/cm² < 0.48 eV
FLUID

CYCLE REQUIREMENTS

RELIABILITY ESTIMATE 99% REQUESTED BY D. A. Brown DATE 10 Sept. 1970

CONFIDENCE LEVEL .95 APPROVED H. L. Springer DATE

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SPECIAL CONSIDERATIONS APPROVED C. E. Dixon DATE

PERTINENT REFERENCES

IDENTIFICATION NO. 619

RADIATION EFFECTS INFORMATION REQUEST

DATE 10 September 1970

SUBJECT AISI 347 STEEL SHEET FUNCTIONAL CATEGORY _____

REF. ITEM SUPPORT NOZZLE, DS-90196 REF. DATA ITEMS C-002, S-110

INFORMATION DESIRED EFFECTS OF FLUENCE AND TEMPERATURE, AT "WORST CASES" IN THE RANGES SHOWN BELOW
(PURPOSE OF STUDY OR EXPERIMENT)
ON THE FATIGUE PROPERTIES.

EXISTING DATA. _____

DATE REQUIRED: Cy 1971

RELATION OF ITEM TO ENGINE COMPONENT OF NOZZLE ASSEMBLY SUBSYSTEM, EC-90151
(LOCATION)

PER DRAWINGS, DATA ITEMS 1138200, 1138000

ENVIRONMENT : NEUTRON FLUX _____ NEUTRON FLUENCE 2.4×10^{19} n/cm²>1.0 MeV
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE 1.5×10^{10} RADS

? TEMPERATURE 40-2060 ° R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER 1.9×10^{18} n/cm²<0.48 eV

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99% REQUESTED BY D.A. Brown DATE 10 Sept. 1970

CONFIDENCE LEVEL .95 APPROVED H. L. Springer DATE _____

APPROVED T. A. Redfield DATE _____

SPECIAL CONSIDERATIONS _____ APPROVED C. E. Dixon DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 620

RADIATION EFFECTS INFORMATION REQUEST

DATE 10 September 1970

SUBJECT HASTELLOY-X FORGING FUNCTIONAL CATEGORY

REF. ITEM SUPPORT NOZZLE, DS-90196 REF. DATA ITEMS C-002, S-110

INFORMATION DESIRED EFFECTS OF FLUENCE AND TEMPERATURE, AT "WORST CASES" IN THE RANGES SHOWN BELOW,
(PURPOSE OF STUDY OR EXPERIMENT)
ON TENSILE PROPERTIES OF PARENT MATERIAL AND WELDMENTS. EVALUATE ANNEALING EFFECTS. INFORMATION SHOULD

SUPPLEMENT EXISTING DATA.*

DATE REQUIRED: cy 1971

RELATION OF ITEM TO ENGINE COMPONENT ON NOZZLE ASSEMBLY SUBSYSTEM, EC-90151
(LOCATION)
PER DRAWINGS, DATA ITEMS 1138200, 1138000

ENVIRONMENT : NEUTRON FLUX NEUTRON FLUENCE 2.4×10^{19} n/cm²>1.0 MeV
(Ref. data items) GAMMA DOSE RATE GAMMA DOSE 1.5×10^{10} RADS
TEMPERATURE 40-800°R NUCLEAR HEATING RATE
PRESSURE OTHER 1.9×10^{18} n/cm²<0.48 eV
FLUID

CYCLE REQUIREMENTS

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SPECIAL CONSIDERATIONS APPROVED C. E. Dixon DATE

PERTINENT REFERENCES

IDENTIFICATION NO. 621

RADIATION EFFECTS INFORMATION REQUEST

DATE 10 September 1970

SUBJECT HASTELLOY-X SHEET

FUNCTIONAL CATEGORY _____

REF. ITEM SUPPORT NOZZLE, DS-90196

REF. DATA ITEMS C-002, S-110

INFORMATION DESIRED DETERMINE EFFECTS OF FLUENCE AND TEMPERATURE AT "WORST CASES" IN THE RANGES SHOWN
(PURPOSE OF STUDY OR EXPERIMENT)
BELOW, INCLUDING INTERACTIONS, ON THE FATIGUE PROPERTIES. EVALUATE ANNEALING EFFECTS.

DATE REQUIRED: CY 1971

RELATION OF ITEM TO ENGINE COMPONENT OF NOZZLE ASSEMBLY SUBSYSTEM, EC-90151
(LOCATION)

PER DRAWINGS, DATA ITEMS 1133200, 1138000

ENVIRONMENT : NEUTRON FLUX _____ NEUTRON FLUENCE 2.4×10^{19} n/cm² > 1.0 MeV
(Ref. data items)

GAMMA DOSE RATE _____ GAMMA DOSE 1.5×10^{10} RADS

TEMPERATURE 40-2060°R NUCLEAR HEATING RATE _____

PRESSURE _____ OTHER 1.9×10^{18} n/cm² < 0.48 eV

FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99%

REQUESTED BY D. A. Brown

DATE 10 September 1970

CONFIDENCE LEVEL .95

APPROVED H. L. Springer

DATE _____

APPROVED T. A. Redfield

DATE _____

SPECIAL CONSIDERATIONS _____

APPROVED C. E. Dixon

DATE _____

PERTINENT REFERENCES _____

IDENTIFICATION NO. 622

RADIATION EFFECTS INFORMATION REQUEST

DATE 10 September 1970

SUBJECT ARMCO 22-13-5 FORGING

FUNCTIONAL CATEGORY _____

REF. ITEM SUPPORT NOZZLE, DS-90196

REF. DATA ITEMS C-002, S-110

INFORMATION DESIRED EFFECTS OF FLUENCE AND TEMPERATURE AT "WORST CASE" CONDITIONS ON TENSILE
(PURPOSE OF STUDY OR EXPERIMENT)
PROPERTIES OF PARENT MATERIAL AND WELDMENTS. EVALUATE THE EFFECT OF ANNEALING.

DATE REQUIRED: CY 71

RELATION OF ITEM TO ENGINE COMPONENT ON NOZZLE ASSEMBLY SUBSYSTEM EC-90151
(LOCATION)

PER DRAWINGS, DATA ITEMS 1138200, 1138000

ENVIRONMENT : NEUTRON FLUX _____ NEUTRON FLUENCE 2.4×10^{19} n/cm²>1.0 MeV
(Ref. data items) GAMMA DOSE RATE _____ GAMMA DOSE 1.5×10^{10} RADS
✓ TEMPERATURE ✓ _____ NUCLEAR HEATING RATE _____
PRESSURE _____ OTHER 1.9×10^{18} n/cm²<0.48 eV
FLUID _____

CYCLE REQUIREMENTS _____

RELIABILITY ESTIMATE 99%

REQUESTED BY D. A. Brown

DATE 10 September 1970

CONFIDENCE LEVEL .95

APPROVED H. L. Springer

DATE _____

APPROVED T. A. Redfield

DATE _____

SPECIAL CONSIDERATIONS _____

APPROVED C. E. Dixon

DATE _____

PERTINENT REFERENCES _____

AEROJET NUCLEAR SYSTEMS COMPANY
SACRAMENTO, CALIFORNIA

TO: C. M. Kawashige 28 January 1971
FROM: T. A. Redfield N4360:MO55:WEF:jb

SUBJECT: Radiation Effects Analysis of AISI 347 Stainless Steel in
the Upper and Lower Ejector Walls of the NASS

COPIES TO: M. J. Davidson, C. E. Dixon, ~~W. E. French~~, R. V. Lichtenberger,
R. L. Murray, T. A. Redfield, R. R. Sternberg, P. P. Ventura, K. Sato

REFERENCE: (a) NASS Memo 71-081, R. V. Lichtenberger to C. M. Kawashige,
dated 28 December 1970, Subj: EDS PDR Package

ENCLOSURE: (1) Preliminary Radiation Effects Analysis of AISI 347
Stainless Steel Walls of the NASS Primary Upper and
Lower Ejectors

Enclosure (1) is herewith submitted for incorporation in the
data package to be submitted for PDR, Reference (a).

T. A. Redfield
T. A. Redfield, Manager
Radiation Effects Section

PRELIMINARY RADIATION EFFECTS ANALYSIS
OF AISI 347 STAINLESS STEEL WALLS OF THE
NASS PRIMARY UPPER AND LOWER EJECTORS

1. SUMMARY AND CONCLUSIONS

An analysis of the available data for 347 stainless steel sheet material specified in the NERVA Altitude Simulation System with respect to the anticipated nuclear environment has been completed.

Various reactor tests have supplied information on tensile properties within the general fluence/temperature range of the NERVA requirements for 347 stainless steel sheet, non-welded, and 347 stainless steel forging, welded and non-welded. While the tests were small in scope and served to provide answers for particular situations not directly comparable to the NASS duct environmental and operating conditions, they did furnish some useful data on the radiation effects to be expected for a minimum of 1000 full power NERVA engine firings of 50 minutes each.

AISI 347 stainless steel sheet and welded sheet can be expected to exhibit decreased ductility at NASS fluences of 9×10^{18} n/cm² (< 0.3 eV) and maximum gas side wall temperatures of 1500°R on the primary ejector upper section and 1275°R on the primary ejector lower section. Evaluation of available data indicated that the decrease in ductility due to radiation will not be significant under NASS duct conditions. At this time no further material characterization tests are considered necessary.

2. DISCUSSION

AISI type 347 stainless steel is the primary candidate material for the NERVA Altitude Simulation System, Print No. 210-3352-S-200. The primary ejector upper section wall is machined "L" channel AISI 347 stainless steel (Ref. 1) and primary lower section wall is welded "L" channel AISI 347 stainless steel.

The estimated nuclear and thermal environments for the primary ejector section walls are as follows:

a. Nuclear (Ref. 2)

Fast neutron flux ($E_n > 1$ MeV)	7×10^{12} n/cm ² sec
---------------------------------------	--

Fast neutron fluences ($E_n > 1$ MeV, 10 hrs at full power)	2.5×10^{17} n/cm ²
---	--

Fast neutron fluences ($E_n > 1$ MeV, cycle life 1,000 full power cycles at 50 minutes each, Ref. 1)	2.1×10^{19} n/cm ²
--	--

Thermal neutron flux ($E_n < 0.4$ eV)	3×10^{12} n/cm ² sec
---	--

Thermal neutron flux ($E_n < 0.4$ eV, 10 hrs at full power)	1×10^{17} n/cm ²
---	--------------------------------------

Thermal neutron fluences ($E_n < 0.4$ eV, cycle life 1,000 full power cycles at 50 minutes each, Ref. 1)	9×10^{18} n/cm ²
--	--------------------------------------

b. Thermal (Ref. 1)

Maximum temperature primary ejector upper section gas side	1500°R
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Maximum temperature primary ejector lower section gas side	1275°R
--	--------

NERVA test experience in the above fluence and temperature ranges for 347 stainless steel sheet is limited to GTR-18 (Ref. 4). This test included 21 specimens (0.012 inch thick sheet, Specification MIL-S-6721B) irradiated to a fluence of 4.2×10^{18} n/cm² (< 0.48 eV), 6.1×10^{17} n/cm² (< 0.48 eV), and 1×10^{19} (< 0.48 eV) and irradiation temperature between 430 and 600°R. The 21 irradiated and 9 non-irradiated control specimens were subsequently tensile tested at 1500, 1750 and 2000°R.

Table 1, AISI 347 Stainless Steel Sheet Mechanical Properties Test Data, is a summary of tensile data obtained for the non-irradiated and irradiated specimens.

Mean values of % elongation, 0.2% yield strength and ultimate strength at 1500°R and fluences of 4.2×10^{18} n/cm² (< 0.48 eV) are summarized below:

	Fluence n/cm ² (< 0.48 eV) -0- 4.2 x 10 ¹⁸	
Elongation, % $\bar{X}$	28.0	28.1
0.2% Yield Strength, ksi $\bar{X}$	24.9	26.8
Ultimate Strength, ksi $\bar{X}$	56.4	57.0

No significant changes in tensile properties due to radiation were observed in the AISI 347 stainless sheet material at this temperature and fluence.

Testing of the remaining irradiated AISI 347 stainless sheet specimens (Table I) in general resulted in increased ultimate and yield strength and decreased % elongation. Figure 1, AISI 347 Stainless-steel Elongation vs Test Temperature for Various Thermal-Neutron Fluences (Ref. 4, Figure 14), summarizes the changes in percent elongation for all forms of 347 stainless steel tested in the GTR-18 test.

Surprisingly, very little irradiation data for 347 stainless steel material is available at the NASS fluences and temperatures. Radiation Effects Information Center, Report No. 45, "The Effects of Neutron Radiation on Structural Materials", contains only two references that are

indirectly applicable to the NASS environment. Table 2, Post Irradiation Mechanical Properties of Type 347 Stainless Steel Plate Specimens Irradiated at Intermediate Temperature (Ref. 5, Table 32), is a summary of tensile data obtained from 347 SS plate irradiated to a fluence of 1×10^{21} n/cm² (fast) in the range 970 to 1210°R and post irradiation tensile tested in the range 540 to 1210°R. This tensile testing resulted in significant increases in tensile strength (ultimate and 0.2% offset yield) and extremely low ductilities (i.e., 1.5% elongation irradiated and 47% elongation unirradiated) when the material was tested either at 540 or 1210°R. However, annealing of the irradiated 347 SS plate at 1460°R for one hour resulted in considerable restoration of the pre-irradiation tensile properties, while annealing at 2260°R for one hour resulted in complete recovery of the pre-irradiation tensile properties except for some ductility. The effect of testing temperature on the post irradiation ductility is illustrated in Figure 2, (Ref. 5, Figure 80). 348 stainless steel, which differs only from 347 SS in that the tantalum content must be less than 0.1 weight percent in type 348, while it could be as high as 10 times the carbon content in type 347, irradiated from 1×10^{20} to 1×10^{21} n/cm² begins to recover in ductility above 1210°R and has almost complete recovery at 1570°R.

No irradiation data are available for welded 347 stainless steel sheet at NASS fluences and temperatures. In GTR-18 (Ref. 4) tensile testing of 347 stainless steel forging weldments (349 SS filler) at 1240 and 1500°R resulted in minor decreases of ultimate and yield strength after

irradiation to a fluence of $4.2 \times 10^{18} \text{ n/cm}^2$ ($< 0.48 \text{ eV}$). Elongation of the 347 forging weldment is increased from 22.2% for the control to 26.8% for specimens irradiated to a fluence of $4.2 \times 10^{18} \text{ n/cm}^2$ ($< 0.48 \text{ eV}$) and tensile tested at 1500°R. The tensile data for 347 stainless steel forgings are summarized in Table 3 and Figure 1 summarizes the changes in percent elongation for all forms of 347 stainless steel tested in the GTR-18 test matrix.

Direct comparison of experimental results from different testing agencies should be avoided due to the variables that affect irradiated tensile properties, namely the composition and form of material, grain size, energy spectrum of the radiation, irradiation temperature, testing temperature and strain rate. However, the above data indicates that 347 stainless steel sheet is an acceptable candidate material for primary ejector upper and lower section walls at the expected maximum NASS fluences and temperatures.

REFERENCES

1. Confirmemo from R. V. Lichtenberger to T. A. Redfield, NASS Duct, dated 18 January 1971.
2. Radiation Levels in the NASS Duct, J. C. Courtney to R. L. Murray, Memo No. 4340:6372M, dated 7 December 1970.
3. Engineering Design Sheet Design and Test Requirements for NASS Exhaust Duct System, S057-DS090221B-EA, June, 1970.
4. Final Report on GTR-18 Test to Determine High-Temperature Radiation Effects on Structural Materials, RN-S-0398, July, 1967.
5. Radiation Effects Information Center Report No. 45, The Effects of Neutron Radiation on Structural Materials, 30 June 1967.

Figure 1 - AISI 347 Stainless-Steel Elongation vs Test Temperature for Various Thermal-Neutron Fluences

FIGURE 2. EFFECT OF TEST TEMPERATURE ON THE DUCTILITY OF IRRADIATED TYPE 348 STAINLESS STEEL IN THE ANNEALED CONDITION

TABLE 1
AISI 347 STAINLESS-STEEL SHEET
MECHANICAL PROPERTIES TEST DATA FROM GTR-18

Test Temperature, °R		1500			1750			2000			
Fluence, nvt (<0.48 eV)		Control	4.2 x 10 ¹⁸	Control	6.1 x 10 ¹⁷	4.2 x 10 ¹⁸	1 x 10 ¹⁹	Control	6.1 x 10 ¹⁷	4.2 x 10 ¹⁸	1 x 10 ¹⁹
Ultimate Strength, ksi	Test Values	55.6	57.5	34.4	37.1	35.3	34.9	11.1	12.1	13.9	11.6
		57.0	56.4	34.0	(35.6)	34.1	34.9	(15.5) ²	11.5	(13.3) ³	12.6
		56.5	(54.6) ¹	36.3	35.5	33.4	35.7	11.3	11.4	12.6	12.2
	Average	56.4	57.0	34.9	36.7	34.3	35.1	11.2	11.7	13.2	12.2
	Range	1.4	1.1	2.3	2.4	1.7	0.8	0.2	0.7	1.3	1.0
	Std. Dev.	0.73	0.78	1.20	1.70	1.00	0.42	0.14	0.37	0.92	0.52
0.2% Yield Strength, ksi	Test Values	25.8	27.0	23.3	22.8	22.6	22.8	10.0	11.0	11.2	9.5
		24.3	26.5	21.5	(31.4)	23.2	22.1	(12.5) ²	10.6	(12.9) ³	11.4
		24.7	(45.9)	22.9	24.4	22.9	22.6	10.3	10.5	12.2	11.2
	Average	24.9	26.8	22.3	23.6	22.9	22.5	10.2	10.8	11.7	10.7
	Range	1.5	0.5	1.8	1.6	0.6	0.7	0.3	0.5	1.0	1.9
	Std. Dev.	0.78	0.35	0.94	1.13	0.31	0.37	0.21	0.26	0.71	1.00
Fracture Strength, ksi	Test Values	54.0	48.0	0	0	0	0	0	0	0	0
		55.0	50.0	0	0	0	0	0	0	0	0
		52.0		0		0	0				0
	Average	53.7	49.0								
	Range	3.0	2.0								
	Std. Dev.	1.6	1.4								
% Uniform Elongation	Test Values	18.7	21.7	11.6	12.7	8.4	12.7	4.7	5.0	4.5	7.0
		21.4	22.3	(18.7)	15.2	12.5	13.3	2.6	3.7	4.0	3.2
		22.2		9.4		11.0	13.0				4.0
	Average	20.8	22.0	10.5	14.0	10.6	13.0	3.6	4.4	4.2	4.7
	Range	3.5	0.6	2.2	2.5	4.1	0.6	2.1	1.3	0.5	3.8
	Std. Dev.	1.83	0.43	1.56	1.77	2.15	0.31	1.49	0.92	0.35	1.93
% Elongation	Test Values	27.9	28.2	38.4	40.4	30.8	35.6	69.5	50.0	44.8	31.9
		26.8	28.0	41.5	(32.0)	33.5	36.9	(66.7) ²	(75.1)	(50.1) ³	38.7
		29.3 ⁴	(28.6)	37.2	39.9	30.7	25.7	68.4	62.7	55.5	30.6
	Average	28.0	28.1	39.0	40.2	31.7	32.7	69.0	56.4	50.1	33.7
	Range	2.5	0.2	4.3	0.5	2.8	11.2	1.1	12.7	10.7	8.1
	Std. Dev.	1.31	0.14	2.25	0.35	1.47	5.86	0.78	9.01	7.59	4.24
% Reduction In Area	Test Values	38.1	35.1	54.1	44.8	32.9	37.0	62.5	48.9	37.1	37.2
		33.1	34.6	55.9	(39.3)	41.2	33.4	(57.2) ²	59.5	(52.4) ³	34.6
		40.5	(34.4)	50.0	53.3	39.3	40.9	58.2	61.9	49.5	32.0
	Average	37.2	34.7	53.3	49.0	37.8	37.1	60.4	56.7	43.3	34.6
	Range	7.4	0.7	5.9	8.5	8.3	7.5	4.3	13.0	12.4	5.2
	Std. Dev.	3.87	0.50	3.09	6.03	4.34	3.93	3.05	6.81	8.79	2.72

NOTE: 1. Values in parentheses were omitted from computations of average, range, and standard deviation.
2. Instron chart data not usable.

3. Low temperature during yield.
4. Calculated from Instron chart.

TABLE 2

POST IRRADIATION MECHANICAL PROPERTIES OF TYPE 304 STAINLESS STEEL PLATE SPECIMENS IRRADIATED AT
INTERMEDIATE TEMPERATURES FROM REIC REPORT NO. 45

Fluence, 10 ²¹ n/cm ²	Irradiation Temperature, °R	Test Temperature, °R	0.2% Offset Yield Strength, 1000 psi		Ultimate Strength, 1000 psi		Elongation, percent				Reduction in Area, percent	
			Unirr.	Irr.	Unirr.	Irr.	Uniform	Total	Unirr.	Irr.	Unirr.	Irr.
1.7	1000	540	56(a)	130	92(a)	134	NA	2.0	57(a)	6.5	65(a)	59.1
1.9	970	540		132		135		1.2		4.3		53.5
3.3	1185	540		143		147		1.0		9.4		61.5
3.9	1085	540		143		145		1.2		6.5		54.5
5.6	1135	540		140		145		1.8		10.6		59.8
7.1	1200	540		143		149		2.0		10.3		58.2
2.1	1000	1210	45(a)	103	75(a)	103		0.6	47(a)	4.5	NA	42.7
2.5	1160	1210		115		115		0.6		4.2		64
3.7	1220	1210		123		123.5		0.7		4.2		45.5
5.1	1200	1210		115		115		0.7		1.5		40.8
6.5	1220	1210		123.6		125.8		0.8		4.1		40.5
6.9	1220	1210		120		120		0.8		3.5		52.5
1.0	1220	540(b)	56(a)	72.3	92(a)	98		20.0	57(a)	44.3	65(a)	56.2
5.3	1195	540(b)		75.0		96		14.0		41.0		63.3
6.5	1220	540(c)		55.8		86.5		25.0		51.0		42.9
6.3	1195	540(c)		36.5		77.5		25.0		41.2		36.9
1.7	1220	1210(b)	45(a)	54.4	75(a)	69.2		11.1	47(a)	16.6	NA	55.8
2.3	1130	1210(b)		58.0		71.4		11.9		19.5		55.2
4.6	1210	1210(c)		33.0		62.5		21.5		26.2		53.6
6.3	1210	1210(c)		27.2		62.5		28.0		41.2		67.3

NA - not available.

(a) Averaged values

(b) Annealed for 1 hour at 1464R

(c) Annealed for 1 hour at 2256R

TABLE 3

AISI 347 STAINLESS STEEL, FORGINGS AND WELDMENTS
MECHANICAL PROPERTIES TEST DATA FROM GTR-18

Material Form		Forging				Weldment			
Test Temperature, °R		1240		1500		1240		1500	
Fluence, nvt ($E < 0.48$ eV)		Control		4.2×10^{18}		Control		4.2×10^{18}	
Ultimate Strength, ksi	Test Values	57.2	57.7	48.9	51.1	59.8	56.2	50.7	46.5
		56.6	53.5	48.4	51.4	56.8	55.3	52.8	52.4
		56.6	53.6		50.1	55.7	54.9	51.6	50.2
	Average	56.8	54.9	48.6	50.9	57.4	55.4	51.6	49.6
	Range	0.6	4.2	0.5	1.3	4.1	1.3	2.1	5.9
	Std. Dev.	0.31	2.20	0.35	0.68	2.15	0.68	1.10	3.09
0.2% Yield Strength, ksi	Test Values	16.6	18.2	(38.3)	17.5	29.2	23.9	26.2	20.0
		16.0	18.5	17.6	21.5	22.9	23.6	26.0	27.8
		20.9	18.1		18.3	22.7	21.2	23.1	22.1
	Average	17.8	18.2	17.6	19.1	24.9	22.8	25.0	23.3
	Range	4.9	0.4		4.0	6.5	2.7	3.1	7.8
	Std. Dev.	2.57	0.21		2.09	3.40	1.41	1.62	4.08
Fracture Strength, ksi	Test Values	45.0	38.0	31.0	36.0	25.0	34.0	36.0	(41.0)
		46.0	34.0	28.0	37.0	34.0	31.0	44.0	24.0
		46.0	40.0		32.0	(50.0)	30.0	(25.0)	20.0
	Average	45.7	37.3	29.5	35.0	29.5	31.7	40.0	22.0
	Range	1.0	6.0	3.0	5.0	9.0	4.0	8.0	4.0
	Std. Dev.	0.5	3.1	2.1	2.6	6.4	2.6	5.7	2.8
% Uniform Elongation	Test Values	24.6	26.0	23.8	23.7	14.5	18.2	14.1	16.2
		25.0	27.5	23.5	23.6	17.0	17.0	14.6	13.7
		25.7	28.2		22.5	16.2	16.6	12.5	18.7
	Average	25.1	27.2	23.6	23.3	15.9	17.3	13.7	16.2
	Range	1.1	2.2	0.3	1.2	2.5	1.6	2.1	5.0
	Std. Dev.	0.58	1.15	0.21	0.63	1.31	0.84	1.10	2.62
% Elongation	Test Values	34.9	34.0	39.8	35.1	22.0	25.5	23.3	28.3
		35.0	38.0	35.3	35.3	25.8	27.2	22.5	22.3
		34.4	37.0		36.3	26.4	26.9	20.7	29.9
	Average	34.7	36.3	37.5	35.6	24.7	26.5	22.2	26.8
	Range	0.6	4.0	4.5	1.2	4.4	1.7	2.6	7.6
	Std. Dev.	0.31	2.09	3.19	0.63	2.30	0.89	1.36	4.00
% Reduction In Area	Test Values	59.4	53.6	62.4	52.7	52.4	59.3	52.0	70.6
		52.6	60.4	55.6	54.3	58.9	61.4	48.5	(34.3)
		60.4	63.8		55.2	61.7	59.8	(66.8)	64.6
	Average	57.5	59.3	59.0	54.1	57.6	60.2	50.2	67.6
	Range	7.8	10.2	6.8	2.5	9.3	2.1	3.5	6.0
	Std. Dev.	4.08	5.34	4.82	1.31	4.87	1.10	2.48	4.26

NOTE: Values in parentheses were omitted from computations of average, range, and standard deviation.